

DEVELOPMENT OF A SAFEGUARDS APPROACH FOR A SMALL GRAPHITE
MODERATED REACTOR AND ASSOCIATED FUEL CYCLE FACILITIES

A Thesis

by

ERIC BENTON RAUCH

Submitted to the Office of Graduate Studies of
Texas A&M University
in partial fulfillment of the requirements for the degree of

MASTER OF SCIENCE

May 2009

Major Subject: Nuclear Engineering

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Approved by:

Chair of Committee,	William S. Charlton
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ABSTRACT

Development of a Safeguards Approach for a Small Graphite Moderated Reactor and
Associated Fuel Cycle Facilities. (May 2009)

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Chair of Advisory Committee: Dr. William S. Charlton

Small graphite-moderated and gas-cooled reactors have been around since the beginning of the atomic age. Though their existence in the past has been associated with nuclear weapons programs, they are capable of being used in civilian power programs. The simpler design constraints associated with this type of reactor would make them ideal for developing nations to bolster their electricity generation and help promote a greater standard of living in those nations. However, the same benefits that make this type of reactor desirable also make it suspicious to the international community as a possible means to shorten that state's nuclear latency. If a safeguards approach could be developed for a fuel cycle featuring one of these reactors, it would ease the tension surrounding their existence and possibly lead to an increased latency through engineered barriers.

The development of this safeguards approach follows a six step procedure. First, the fuel cycle was analyzed for the types of facilities found in it and how nuclear material flows between facilities. The goals of the safeguards system were established next, using the normal IAEA standards for the non-detection and false alarm probabilities. The 5 MWe Reactor was modeled for both plutonium production and maximum power capacity. Each facility was analyzed for material throughput and the processes that occur in each facility were researched. Through those processes, diversion pathways were developed to test the proposed safeguards system. Finally, each facility was divided into material balance areas and a traditional nuclear material accountancy system was set up to meet the established safeguards goals for the facility.

The DPRK weapons program is a great example of the type of fuel cycle that is the problem. The three major facilities in the fuel cycle, the Fuel Fabrication Facility, the 5 MWe Reactor, and the Radiochemical Laboratory, can achieve the two goals of safeguards using traditional methods. Each facility can be adequately safeguarded using methods and practices that are relatively inexpensive and can obtain material balance periods close to the timeliness limits set forth by the IAEA. The Fuel Fabrication Facility can be safeguarded at both its current needed capacity and its full design capacity using inexpensive measurements. The material balance period needed for both capacities are reasonable. For the 5 MWe reactor, plutonium production is simulated to be 6.7 kg per year and is on the high side of estimates. The Radiochemical Laboratory can also be safeguarded at its current capacity. In fact, the timeliness goal for the facility dictates what the material balance period must be for the chosen set of detectors which make it very reasonable.

DEDICATION

This work is dedicated to my parents, Marshall and Julia Rauch, and my brother, Jared Rauch, for always pushing me to give my best in everything I do. Thank you for always being my biggest supporters, and I love you very much.

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Finally thanks to my wonderful wife, Jessica. I know you put up with a lot of absences, but you never complained, thank you.

NOMENCLATURE

IAEA	International Atomic Energy Agency
DPRK	Democratic People's Republic of Korea
MUF	Material Unaccounted For
ISIS	Institute for Science and International Security
PUREX	Plutonium (PU) Uranium (U) Recovery (R) Extraction (EX)
NPT	Nuclear Non-Proliferation Treaty
INFCIRC	IAEA Information Circular
KEDO	Korean Peninsula Economic Development Organization
MBA	Material Balance Area
MBP	Material Balance Period

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CHAPTER I

INTRODUCTION

A. Introduction

Small graphite-moderated and gas-cooled reactors have posed a proliferation risk for many years.⁽¹⁾ These reactors have been around since the beginning of the atomic age and are direct descendents from the first reactor ever built: the Chicago Pile-1 designed by Enrico Fermi during the Manhattan Project. Besides the Calder Hall style reactors in the United Kingdom and the Advanced Gas-Cooled Reactor systems in Germany, this reactor type has primarily been used for plutonium production for military purposes. Due to this fact, there has been little effort put forth to devise a system of safeguards for this reactor system and the international community has avoided building them because their existence would inherently suggest a nuclear weapons program. These reactors are relatively easy to build, have simplified engineering needs, can use natural uranium as fuel and normally have low burnup which allows them to be almost perfect candidates for a covert plutonium production program.

There are, however, several peaceful and legitimate purposes for these systems if they could be properly and responsibly safeguarded by the International Atomic Energy Agency (IAEA). As part of a nation's electrical power strategy, these small reactors could be used to better the standard of living in developing countries as well as provide cheap, reliable electricity and heat to rural communities. As with most technologies that start out in a weapons development project, the potential for misuse is present, with the most notable example of misuse coming from the Democratic People's Republic of Korea (DPRK). However, there are no technical barriers that prevent such a system from being reliably safeguarded under either an INFCIRC-153 or INFCIRC-540 type agreement.^{(2) (3)} If a safeguards approach for a small graphite-moderated and gas-cooled

This thesis follows the style of *Nuclear Technology*.

reactor with associated fuel cycle facilities could be developed and demonstrated to meet the timeliness goals of the IAEA it could allow the IAEA to focus better on those areas requiring greatest attention.

A fuel cycle featuring a graphite moderated reactor is likely to be more common in the future because a few of the designs being considered for the next generation of reactors are graphite moderated.⁽⁴⁾ They have some key benefits that currently marketed light-water reactors do not have. These benefits include lower power output which could be suitable for less developed electrical grids, simpler engineering and material needs, and the ability to forego an enrichment program to produce fuel for the reactors. However, these reactors typically have low maximum burnups which results in producing plutonium more suitable for weapons fabrication than traditional light water reactors. For weapons purposes, having a high ^{239}Pu to ^{240}Pu ratio is desired due to the larger probability of fission per capture in ^{239}Pu and a high spontaneous fission rate in ^{240}Pu . ^{239}Pu is produced by the absorption of a neutron in ^{238}U which will beta decay twice to ^{239}Pu . ^{240}Pu is produced by a subsequent neutron absorption in ^{239}Pu through a radiative capture reaction. A general build up and decay scheme for ^{238}U is shown in Fig. 1. As the fuel continues to be irradiated, more ^{239}Pu is produced from ^{238}U ; however, the rate of ^{240}Pu production increases faster than the rate of ^{239}Pu production. Therefore, at higher burnups the ^{240}Pu content grows to what might be unacceptably high contents for weapons purposes. Thus, systems that are well suited for plutonium production are those with high loadings of ^{238}U and low discharge burnups. Both of these facts are present in the operating conditions found in graphite-moderated reactors.

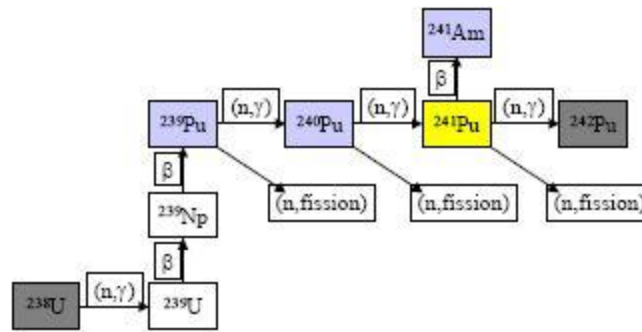


Fig. 1: Production of Plutonium from Uranium-238

Unlike light-water moderated reactors, graphite-moderated reactors can reach criticality using natural uranium which is over 99% ^{238}U .⁽⁵⁾ Natural uranium fuel is normally limited to very low burnups due to criticality constraints. Combined, those 2 facts make any graphite-moderated reactor program suspicious to the international community.

To alleviate concerns in the international community over programs involving graphite moderated reactors, steps could be taken by a state to lengthen their nuclear latency. Nuclear latency can be defined as the effective time required by a state to produce a conventionally deliverable nuclear explosive device. One step that would increase a state's latency and increase the probability of detection of proliferation is the implementation of a rigorous safeguards regime.

B. General Nuclear Fuel Cycles

Most fuel cycles require 3-4 stages, depending on the goals that the fuel cycle is designed to meet. First, raw material must be processed into a suitable fuel. Depending on the type of reactor, the amount of ^{235}U might have to be enriched to achieve a critical system. That uranium is then formed into a chemical composition that is specific to the type of reactor it will be burned in. That uranium is then formed into the fuel shape, normally rods or pellets, and encased in a cladding. The fuel is then burned in the reactor, where uranium is fissioned to release thermal energy. Each fission also releases

a few neutrons which cause further fissions as well as transmutation of other isotopes inside the core.⁽⁶⁾ The most important reaction for safeguards designers is the production of ^{239}Pu through the absorption of a neutron in ^{238}U and two subsequent beta decays. The uranium fuel is burned in the reactor for a certain amount of time, which causes the fuel to have a specific burnup. Burnup is a term that describes how much energy has been released per mass of fuel and is normally given in the units of megawatt-thermal-days per metric ton of initial heavy metal, in most reactors the heavy metal is uranium.⁽⁷⁾ The fuel is removed from the reactor and allowed to cool. About 6% of the total core thermal power is still being produced in the spent fuel due to fission product decay.⁽⁸⁾ Most fuel cycles allow the fuel to cool in a pool of water for a varying amount of time from a few months to a few years. Some fuel cycles then store this spent fuel as is, without reprocessing it. However, most of the fuel cycles in the world do include reprocessing of the spent fuel to recover plutonium and/or uranium for future uses. In this reprocessing, the fuel is most often chemically separated into 3 or more streams, one that produces separated and purified uranium, another one with separated and purified plutonium, and the last is everything else which includes fission products and trans uranic elements. The Pu-U-Recovery-EXtraction (PUREX) process is the most commonly used process in the world for reprocessing.

After the fuel cycle has been properly mapped, the throughput of each facility must be analyzed to determine the material flow through the cycle. The amount, form, and chemical transformations of the material is a very important piece of information needed to adequately design safeguards to detect diversion of material. There are three stages in the fuel cycle: fuel fabrication, transmutation, and post-transmutation. Each stage could be made up of several facilities depending on the fuel cycle. For example, the fuel fabrication stage could consist of mining and milling facilities, an enrichment facility, and a fuel fabrication facility. But not all fuel cycles require an enrichment facility, so the constituent facilities can change based on the needs and design of the entire fuel cycle.

C. The History of Applying Safeguards

Shortly after World War II and the destruction of Hiroshima and Nagasaki, the non-proliferation movement began. It was evident that weapons with such power must be controlled and the ability to produce them must be carefully monitored. Beginning in the 1950's, the first steps toward stopping other countries from developing the technology to produce nuclear weapons were made. The IAEA was formed by the United Nations (UN) in 1957 and was tasked to safeguard nuclear materials in facilities that were placed under its monitoring by member nations.⁽⁹⁾ The scope of IAEA agreements was much smaller at first, as any inspections conducted by the new agency had to have a 1 week warning. Only reactors above 100 MW(th) would be subject to safeguards.⁽⁹⁾ Eventually, the first step towards fuel cycle safeguards was taken with INFCIRC-66/rev.2 in 1967, which included provisions for all of the common fuel cycle facilities: fuel fabrication, reactors of all sizes, and reprocessing facilities.⁽¹⁰⁾ With this model in place, the attention of the international community became focused on the Non-Proliferation Treaty (NPT), which is now the single most adopted treaty in the world.

The NPT was an agreement that limited the spread of nuclear weapons among member nations with the promise that those who forego weapons programs will benefit from the technical expertise of the weapons states in building civilian nuclear energy infrastructure. The non-weapons states must complete a safeguards agreement with the IAEA after signing the treaty that will allow for transparency in the fuel cycles of member states. The weapons states are also expected to eliminate their own nuclear weapons in the future. The weapons states are the 5 nations who had nuclear weapons prior to the date of enforcement in 1970: the United States, United Kingdom, Soviet Union (now the obligations are assumed by Russia), People's Republic of China, and France.

The model proposed in INFCIRC-66/rev.2 provided the basic foundations used for the next quarter century. Eventually, uranium enrichment facilities were also

covered under safeguards agreements, as that type of facility was brought online in non-weapons states. The tenets of INFCIRC-66/rev.2 were folded into INFCIRC-153 in 1972 to become a package of safeguards on all nuclear material instead of a menu of potential safeguards, which INFCIRC-66/rev.2 was.⁽²⁾

INFCIRC-153 stood as the primary safeguards approach until after the first Gulf War in 1991, when Iraq's clandestine nuclear weapons program came to light. The extent of the program had been vastly underestimated, and the ability of the IAEA to guarantee that no nuclear material was being used in a weapons program was put in doubt. The next year, the DPRK submitted its initial declaration to the IAEA. The IAEA had difficulty confirming the completeness and the correctness of the declaration.⁽¹¹⁾ Both of these crises showed flaws in the IAEA system for detecting covert nuclear programs. The IAEA sought to increase its effectiveness INFCIRC-540, more commonly called the Model Protocol or Additional Protocol.⁽³⁾ The Additional Protocol had provisions for the IAEA to demand special inspections of facilities not covered in a state's declaration, broadened the right of the IAEA to order surprise inspections, and allowed member nations to share intelligence about other state's programs with the IAEA in an effort to stop the types of programs in Iraq and the DPRK before they became so advanced. The Additional Protocol signified a significant shift in the IAEA's dealing with member nations. Instead of verifying only declared activities, the IAEA would now be trying to detect any covert nuclear activities, even those completely separate from a declared program.⁽⁹⁾ The Additional Protocol is the current set of goals the IAEA operates under. Any new safeguards agreement with a country will be of INFCIRC-540 type and will include the right of the IAEA to inspect facilities not declared by the country.

D. Nuclear Safeguards

Nuclear material safeguards is an internationally recognized and implemented system for detecting and deterring proliferation by states without nuclear weapons. The objectives of any IAEA safeguards system are twofold:

1. Timely detection of a diversion of a significant quantity of special nuclear material from a declared facility.
2. Timely detection of undeclared production of special nuclear material from undeclared facilities.

There are six major steps in developing a safeguards regime.

1. Research the fuel cycle in question to understand what facilities are included and what the relationships between facilities are.
2. Establish the safeguards goals for the fuel cycle.
3. Model the reactor(s) to determine their plutonium production and uranium consumption as a function of burnup and to determine their maximum safe power.
4. Analyze the facilities in the fuel cycle for material throughput and to understand the types of processes that occur at each facility.
5. Assess proliferation pathways through each facility and determine where a measurement could detect diversion. Design a system of measurement devices that will detect diversion of a significant quantity of material in a timely manner.
6. Perform an uncertainty analysis on the proposed system of measurements to determine the uncertainty in the material unaccounted for (MUF) and that they system can reliably meet the defined safeguards goals.

These six steps allow for the generation of a safeguards regime that is specific to a particular fuel cycle.

The first step in designing any safeguards regime is to determine what facilities are currently in the fuel cycle, what their responsibilities are to the fuel cycle and what their throughput is. Typically this information would be reported by the state to the

IAEA; however it can also be obtained through many sources, including non-governmental entities [like the Institute for Science and International Security (ISIS)], the IAEA, or reports generated for the United States Government. It is important to generate as full an understanding as possible about the way in which the facilities that make up the fuel cycle process and produce fissile material.

The next step in designing a safeguards regime is to state the goals for the regime. The goals are normally defined in terms of a goal quantity, a timeliness goal for the detection of a goal quantity sized diversion, a non-detection probability, and a false alarm probability. The IAEA has defined the first two and they are broken down into categories. For the goal quantity, the IAEA has defined a set of Significant Quantities (SQ) based on the type of material in question. These values are derived from the amount of material that would probably be needed by any state to make a nuclear device:

- 1 SQ of material =
 - 8 kg of Pu or ^{233}U
 - 25 kg of ^{235}U in HEU (^{235}U enrichment > 20%)
 - 75 kg of ^{235}U in LEU/NU/DU (^{235}U enrichment < 20%).

The timeliness goals are also broken down into categories based off of the time it would take to form the material into a device:

- Timeliness Goal =
 - Un-irradiated, Direct Use material = 1 month
 - Irradiated, Direct Use material = 3 months
 - Indirect Use material = 1 year

Direct use materials are also defined as materials that are suitable for weapons making purposes like Pu, ^{233}U , or HEU. Indirect use materials are defined as materials that can be used to create direct use materials like LEU, NU, and DU. ⁽¹²⁾

The false alarm probability (α) and the non-detection probability (β) are derived from a statistical analysis of a measurement system. α is defined as the probability that the system will alarm given that no diversion has occurred and β is the probability that the system will not alarm given that a diversion has occurred. Any measurement will

have an uncertainty associated with it. That uncertainty can be assumed to have a normal distribution around the true value of the measurement. In the design of the safeguards regime, α and β are set at a value determined by the designer. In Fig. 2, a threshold, S , is selected where any value above that threshold will alarm the system and any value below will not. α is the area under the No Diversion curve to the right of the alarm threshold, and β is the area under the Diversion curve to the left of the alarm threshold. The threshold, S , can be moved to achieve different α 's and β 's. Also, the uncertainties of the measurements can be changed through the use of different detectors, which would cause the curves to change their standard deviation which would change α and β at the same threshold.⁽¹³⁾

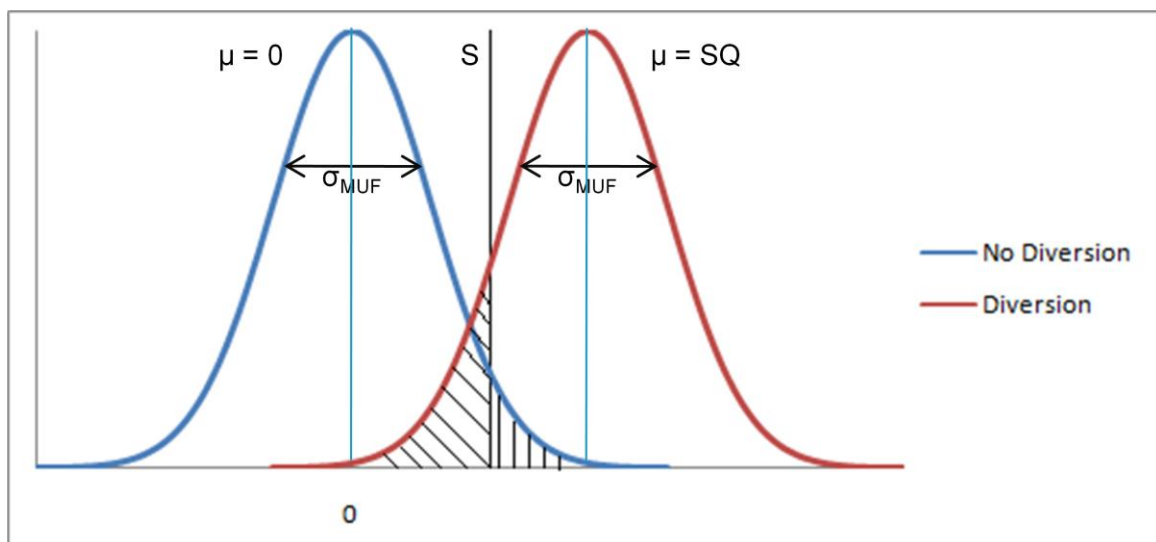


Fig. 2: Selecting the Alarm Threshold

By selecting α and β , the alarm threshold can be set. The goals the IAEA has set for most safeguards regimes 5% for both α and β . This can be translated into an amount of uncertainty that the MUF can have to achieve the stated safeguards goals. That amount for α and β set to 5% is $3\sigma_{\text{MUF}} < 1 \text{ SQ}$ of material. Therefore, the alarm threshold is normally set for 1/3 of the SQ material in question.

Concurrently with the analysis of the fuel cycle facilities will be modeling of the reactors to get an estimate of the plutonium production and overall performance of the

reactor. Most reactor analysis codes do a fair job of analyzing the reactor neutronics, and they can be used to estimate the plutonium production of a particular reactor. However, that estimate is normally $\pm 5\text{-}25\%$, which means large fuel cycles cannot use this estimate in a material accountancy program.⁽¹⁴⁾ It is also important to model the thermodynamic cycle of the reactor to understand what its thermal limits are. Plutonium production is influenced by reactor power which is primarily limited by the materials used in the construction of the reactor. The main issues to consider are the melting of the fuel and failing temperature of the cladding, though other safety characteristics like departure of nucleate boiling could be used as a limiting factor depending on reactor type.⁽⁸⁾

After the fuel cycle has been mapped and each facility inside it has been examined for throughput and process, the facilities in the fuel cycle must be analyzed for proliferation pathways. Proliferation pathways are defined as the means by which fissile material can be taken out of the fuel cycle and used for other purposes. The methods and targeted material will change at each stop in the fuel cycle, but the goal is the same, to determine where the material is at the most risk for being diverted. Once these pathways have been analyzed, the fuel cycle should be analyzed for key points where a particular measurement can detect a diversion of a significant quantity of material. These key measurement points will make up the basis of the safeguards approach.

The system of measurements will have an uncertainty associated with it. By knowing the throughput of material and the uncertainty of the measurements within the system, an amount of MUF can be calculated. The uncertainty of the MUF should meet the IAEA's timeliness goals for fissile material.

E. Overview of Work

The expansion of nuclear power will spread both the benefits and problems associated with its use, and in order to enjoy the benefits a state must be ready and willing to tackle the problems that nuclear energy will present. Beyond the technical

challenges associated with developing a nuclear energy infrastructure, the special circumstances regarding dual use technologies and their consequences if misused or abused by malefactors require vigilance by the international community to deter and detect any such misuse. The next generation of reactors that will power many countries through the 21st century will feature designs and materials that have not been popular previously. Graphite moderated reactors will likely be part of the new generation, and therefore work must be done now to prepare for those reactors coming online.

As reactors spread to more nations and become more abundant, the need for assurances that this technology cannot be used for weapons purposes grows. Successfully proving that a small to medium sized fuel cycle featuring a graphite moderated reactor can be safeguarded would open up their use to developing nations. In order to prove that safeguarding such a fuel cycle is possible, the first step is to choose a test case that can be adequately modeled and simulated. In the simulation, add the appropriate safeguards and determine if the traditional method of safeguards implementation will meet the traditional safeguards goals. For this work, the fuel cycle at the Yongbyon complex in the DPRK was chosen.

There are several reasons why the DPRK fuel cycle was chosen. First, it features three key facilities present in most fuel cycles: fuel fabrication, reactor, and reprocessing. All of these facilities are co-located in Yongbyon. The fuel fabrication and reprocessing facilities were designed to accommodate a much larger fuel cycle, which allows for larger capacity facilities to also be modeled through this fuel cycle. The fuel cycle's of developing countries would also share many characteristics of this fuel cycle; such as overall power output, amount of material needed to sustain operation, and amount of production of special nuclear material. Finally, the open source literature on this fuel cycle contains many details that further increase the accuracy of the model as well as serves as a check on the data the model provides.

CHAPTER II

HISTORY AND TECHNICAL DETAILS OF THE MAJOR FACILITIES AT YONGBYON

The Yongbyon complex is a series of facilities that comprise the bulk of the nuclear infrastructure inside of the DPRK. This complex is the primary research site for the entire nuclear program for the DPRK. There are three major facilities located here, and these are the three facilities that are the focus of the test case. Those facilities are the Fuel Fabrication Facility, the 5 MWe Reactor, and the Radiochemical Laboratory. The Yongbyon complex is located approximately 100 km north of Pyongyang.⁽¹¹⁾ Fig. 3 and Fig. 4 are a map of Yongbyon and a satellite photo, respectively. As can be seen, the complex is intertwined with the Kuryong River. The Yongbyon complex forms a reasonably self-contained fuel cycle with both a front-end and back-end. Thus, it is a useful test case for exploring safeguards approaches for generally collocated facilities. The history and technical details of these facilities will be given in this chapter.

The planned purpose for the Yongbyon complex was to provide support for a series of production reactors. The 5 MWe Reactor was built first as a test reactor and for instruction on reactor operation. The Fuel Fabrication Facility and Radiochemical Laboratory were constructed to support the other planned reactors, and due to that, they both have higher capacities than the needs of the 5 MWe Reactor alone. Two more reactors, named the 50 MWe Reactor and the 200 MWe Reactor, were planned but never finished. These reactors would have been designed similar to the 5 MWe, with larger sizes to produce larger power output. Fig. 2 shows the location of the 50 MWe reactor site, but the 200 MWe reactor was planned for another location outside of Yongbyon. Fig. 2 also shows the location of the reactor supplied by the former USSR, the IRT-2000.

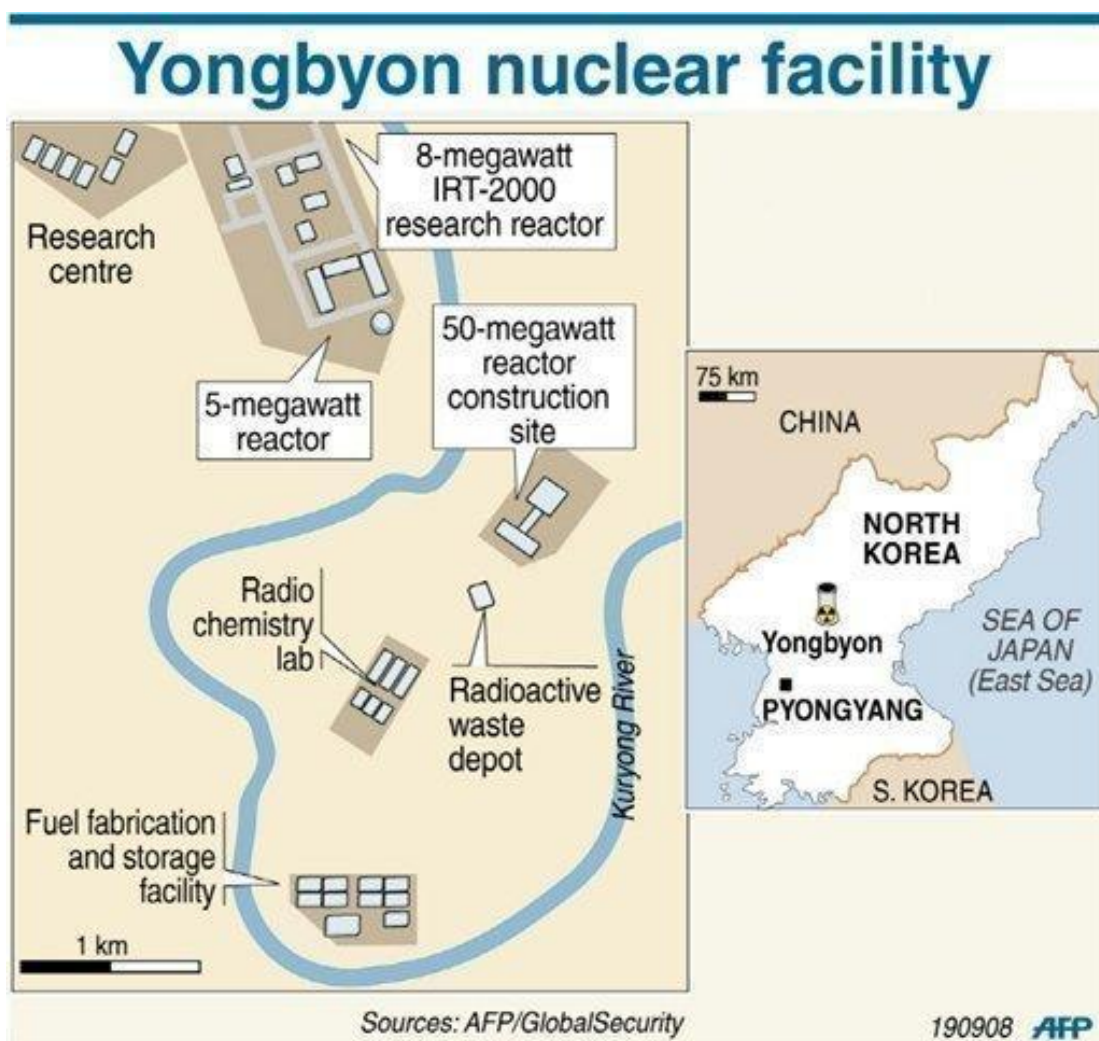


Fig. 3: Map of Yongbyon Facility



Fig. 4: Satellite View of Yongbyon

A. The History of the DPRK Weapons Program

The Democratic People's Republic of Korea has been pursuing the capability of producing nuclear weapons since the 1960's. The first efforts in this program were started under the supervision of the U.S.S.R. which provided a small research reactor in 1965 under a nuclear cooperation treaty signed in 1959. This reactor, known as the

IRT-2M (not shown on photo), was installed as a 2 MW(th) research reactor and was placed at the Yongbyon Nuclear Research Facility. The IRT was originally fueled by 10% enriched fuel, but in 1974 was modified to use 80% enriched fuel and produce 4 MW(th) and later to 8 MW(th) in the 1980's.⁽¹¹⁾ The DPRK joined the IAEA in 1974 and the IRT was then placed under IAEA safeguards in 1977 after being pressured from the Soviet Union. The IRT was the training reactor with which North Korean scientists learned how to operate a reactor. Under supervision from Chinese nuclear specialists, North Korean scientists began to learn how to manage a reactor and the facilities associated with it.

Beginning in 1975, the first steps toward a full weapons program were initiated with the Isotope Production Facility. This facility successfully reprocessed natural uranium targets irradiated in the IRT and produced approximately 300 mg of plutonium according to the declaration given by the DPRK.⁽¹¹⁾ The Isotope Production Facility was a lab-scale reprocessing facility designed to test the PUREX method for obtaining plutonium from spent nuclear fuel. It consisted of a handful of hot-cells. This capability gave the DPRK experience in both operating reactors and reprocessing spent fuel. The plan for a much larger fuel cycle began to take shape.

The planned fuel cycle has everything that would be needed for production of plutonium through the use of 3 reactors. The Fuel Fabrication Facility was designed to accommodate the 5, 50, and 200 MWe Reactors, and the Radiochemical Laboratory able to accommodate at least the 5 and 50 MWe Reactors. The components of that plan are shown in Fig. 5.

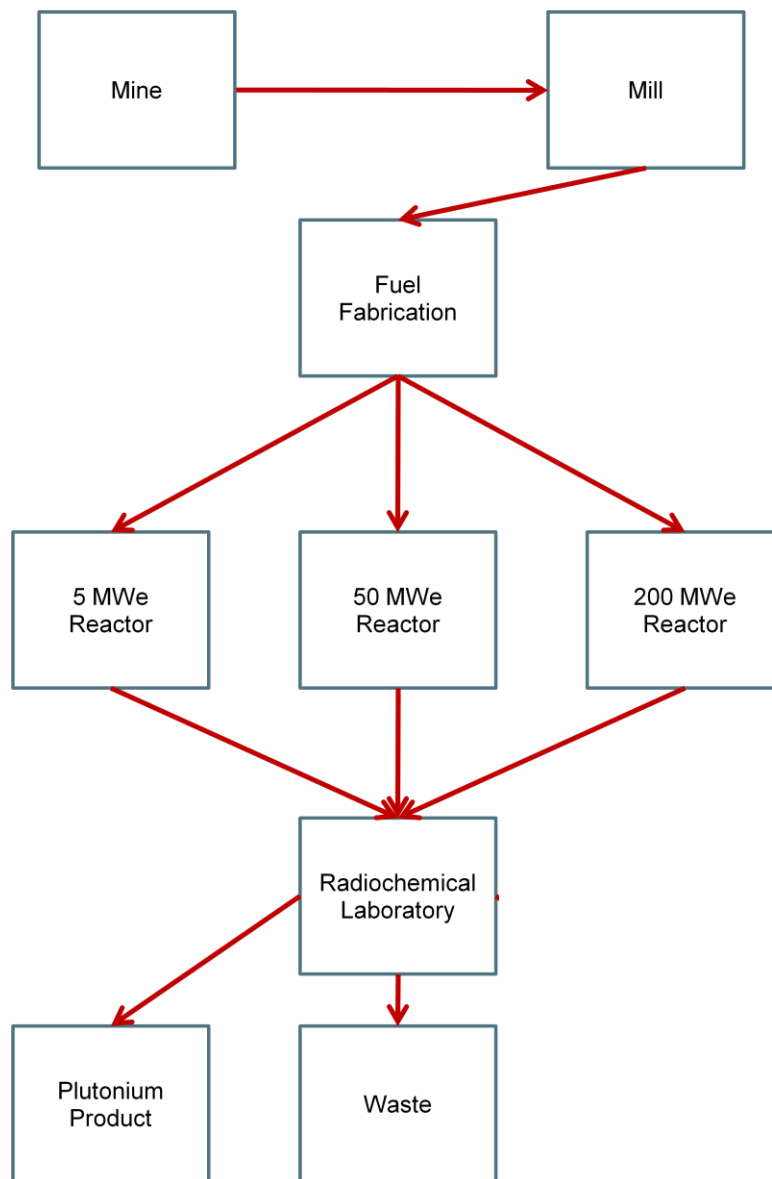


Fig. 5: Planned Fuel Cycle

Construction began on the 5 MWe Reactor in 1980, and it was completed in 1986.⁽¹¹⁾ The design of the reactor is similar to the Calder Hall reactor of Great Britain. This Magnox-style reactor is graphite-moderated, CO₂ cooled, and designed for a maximum of 20 MW(th) output.⁽¹¹⁾ It uses natural uranium, so no enrichment capabilities are needed to produce fuel for it indigenously. The specifications for the reactor core, fuel rods, and control rods can be found in Table 1.

Table 1: 5 MWe Characteristics

Thermal Power	20	MWth
Electrical capacity	5	MWe
Initial U Loading	50	MTU
Fuel Channels	812	
Fuel Rods per Channel	10	
Channel Center Pitch	20	cm
Effective Core Height	590	cm
Effective Core Diameter	643	cm
Diameter of Fuel Channel	6.5	cm
Fuel Composition	U - 0.5% Al	
U Slug Diameter	2.9	cm
U Slug Length	52	cm
U Slug Mass	6.242	kg
Cladding and Fin Composition	Mg - 0.5% Zr	
Cladding Outside Diameter	3	cm
Fin Diameter	5	cm
Overall Length one Fuel Pin	60	cm
Number of Control Rod Channels	44	
Diameter of Control Rod Channel	6.5	cm
Control Rod Composition	B ₄ C	
Control Rod Length	520	cm
Coolant	CO ₂	
Reactor Inlet Temp	200	C
Reactor Outlet Temp	360	C

A full scale reprocessing plant was then constructed, giving the Yongbyon facility an industrial size facility to reprocess all of the spent fuel from the 5 MWe and the planned 50 and 200 MWe reactors. A fuel fabrication plant capable of producing enough fuel for both the 5 MWe and 50 MWe annually was also constructed during this time.

A.1. Standoff and Special Inspections

After the DPRK signed the NPT in 1985 at the behest of the USSR, it produced a declaration of its nuclear facilities and previous activities in 1992.⁽¹⁵⁾ Normally reports are due to the IAEA within 18 months of signing the NPT; however, the DPRK took longer. In the report, the DPRK claimed to have separated less than 100g of plutonium from ruptured fuel rods removed from the 5 MWe reactor along with around 300 mg from the IRT. The declaration said that 76 rods were taken from the 5 MWe reactor and replaced. Those spent fuel rods were added to 172 fresh fuel rods to run a hot test of the Radiochemical Laboratory. From this hot test, the DPRK declared it recovered about 62g of plutonium with about 2.4% ^{240}Pu content. They had calculated that the rods had around 90g of plutonium within them before the campaign, and the difference was considered as lost.⁽¹¹⁾

Inconsistencies started to emerge after the IAEA analyzed swipe samples from around the Radiochemical Laboratory and other samples taken from the declared waste and product. The declared waste from each process in the PUREX cycle was analyzed for ^{240}Pu content. The ^{240}Pu content in the plutonium residue in the waste stream of the dissolution of the fuel and cladding was found to be 2.27%. The waste stream from the fission product removal process was found to have a 2.21% ^{240}Pu content. A 2.25% ^{240}Pu content was found in the waste from the U/Pu extraction and a 1.31% ^{240}Pu content was found in the plutonium extraction waste. The solvent recovery waste had a 1.63% ^{240}Pu content. The disparity between the ^{240}Pu content in all of these waste streams and the product indicates that the declaration is either incomplete or incorrect or both. Each ^{240}Pu content sampled matches with some part of the declaration. The product was found to be 2.44% ^{240}Pu , but if that was the case the true plutonium content in those fuel rods should have been around 150g. To produce 90g of plutonium in the spent fuel, the fuel could not be burned as long and it would produce a ^{240}Pu content of around 1.4%, which is closer to the lower ^{240}Pu contents measured.⁽¹¹⁾

An analysis of the dates of reprocessing was also performed. This is done by measuring the amount of ^{241}Am present in the sample, which is only produced by ^{241}Pu beta decay. It is a very accurate way of measuring the amount of time that has passed since the purification of the plutonium product. The DPRK only declared one campaign that occurred in 1989. The ^{241}Am analysis of the different swipe samples showed other campaigns in 1989, 1990, and 1991.⁽¹¹⁾ Further analysis of the waste streams also showed that spent fuel was sent through the reprocessing facility without fresh fuel with it. This is another contradiction to the declaration.

Hans Blix, then director of IAEA, ordered special inspections in 1993 to clear up some of these inconsistencies. As a result, the DPRK announced it was withdrawing from the NPT under Article X, which allows member nations to do so in the name of 'supreme national interests' 90 days after they declare withdrawal. In April of 1994, the 5 MWe was shut down for a refueling operation. Former President Jimmy Carter went to the DPRK to seek some sort of agreement to halt the program during this time. A freeze was put in place in response to this, but Kim Il Sung, the leader of the DPRK since the country's foundation, died in July of 1994. Power was transferred to one of his sons, Kim Jong-Il, and the negotiations continued for an enduring agreement. In August of 1994, the Agreed Framework was agreed to by all parties.⁽¹¹⁾

A.2 Agreed Framework

The Agreed Framework allowed for an incentive based roll-back of the program. Different benchmarks would be met with continued aid and rewards. In essence, the DPRK would dismantle the 5 MWe reactor and the Radiochemical Laboratory, halt construction of the 50 MWe and 200 MWe reactors, store all spent nuclear fuel, continue to be a signatory of the NPT in full compliance, and implement the denuclearization agreement it had signed with the Republic of Korea. In return, the United States would provide, through the Korean Peninsula Energy Development Organization (KEDO), the monetary assistance to provide 500,000 tons of heavy fuel oil annually and two light

water reactors. The conditions for certain deliveries were based on compliance with the IAEA and the DPRK's NPT obligations.⁽¹⁶⁾ Fuel oil shipments began almost immediately.

The fuel discharged from the 5 MWe Reactor would normally corrode under water, and observers noted that the spent fuel pond on site was cloudy and had inadequate filtering equipment.⁽¹⁷⁾ Steps were taken to seal and store the spent fuel through a canning system which would store the normally very easily corrodible magnesium cladding inside a can with an inert atmosphere. These cans could then be stored underwater without fear of radioactive release. Construction was also begun on the two LWR's. However, the agreement started to deteriorate. KEDO was the lead actor in implementing the Agreed Framework. It was in charge of the fuel oil deliveries and the construction of the two LWR's. Due to the political nature of the region and its underlying tension, there were several factors that hindered its administration. These factors ranged from requiring a consensus from each of the Executive Board members to do anything to being bound to the prevailing political climate of the Korean Peninsula and Northeast Asia as a whole. Therefore, other factors besides those directly involved with the mission of KEDO affected its performance.⁽¹¹⁾

The DPRK also slowed down implementation of the Agreed Framework. The continued design and testing work on ballistic missiles worried some in the United States and the other KEDO partners; South Korea, Japan, and the European Union (joined in 1997). The slow movement to dismantle the 5 MWe reactor and the Radiochemical Laboratory also delayed deliveries of both heavy fuel oil and critical parts for the two LWR's. Combined with the difficulties facing KEDO, the untrusting atmosphere around the entire situation led to the end of the Agreed Framework.⁽¹¹⁾

A.3. Restart and Test

In 2002, the Agreed Framework was abandoned and North Korea began reprocessing the 8000 fuel rods removed from the 5 MWe reactor in 1994. In early

2003, the 5 MWe reactor was refueled and restarted, though there was no evidence of restarting construction at the 50 MWe and 200 MWe reactor sites. The 5 MWe reactor was run continuously until 2005, where it was shut down for 45 days for refueling, and another 8000 fuel rods were sent to be reprocessed. The reactor was refueled with fresh fuel to continue production of plutonium.

On October 9, 2006, the DPRK tested a nuclear device. Initial reports suggested a yield between 5 and 15 kt. Later calculations from test data showed that the test was less than 1 kt in yield. Fission product gasses were found in the atmosphere in the days after the test, confirming that a nuclear detonation had occurred. There are varying opinions regarding the reason for such a small test, the most prominent being that this was a failed test. In the days after that first test, there were reports of preparations for subsequent tests; however, no other tests have been confirmed.

The test was internationally condemned and prompted a new interest in the Six-Party Talks. The Six-Party Talks resumed on October 31, 2006. On February 13, 2007, the Six-Party Talks finally came to an agreement to begin the process to permanently end and irrevocably disable the Yongbyon complex. The plan called for 12 disablement steps, which could be completed in any order, and ranged from defueling the 5 MWe Reactor to removing key pipelines in the Radiochemical Laboratory. The 5 MWe Reactor was shut down again in July of 2007 as part of the agreement achieved by the Six-Party Talks.⁽¹⁸⁾

A.4. Decontamination and Dismantlement

It is the current policy of the United States that the entirety of the DPRK nuclear program be dismantled. This policy could require disposal of completed nuclear devices. In the past, 2 different methods have been used in this situation. The first method would be a simultaneous disposal and verification similar to the methods used to verify arms reduction treaties between the US and the USSR/Russia. The second

method would be to verify the disposal of the weapons after their dismantlement similar to the situation of South Africa, where the weapons were destroyed before they were acknowledged. Either option requires access to sensitive information by the IAEA and meticulous record-keeping.⁽¹⁹⁾

The 5 MWe Reactor was shut down in July of 2007 as part of the agreement reached in the Six-Party Talks in February of 2007. The dismantlement of the Radiochemical Laboratory has progressed and would take some time to reverse. As of February 2008, 10 of the 12 disablement steps had been completed.⁽¹⁸⁾ The 5 MWe is in the process of being defueled currently, at the rate of about 20 rods per day. The cooling tower was destroyed by demolition under the watch of international media personnel on June 27, 2008. In the newest declaration of facilities, inventory, and activities, the DPRK admitted to having roughly 30 kg of separated plutonium.⁽²⁰⁾ This is on the lower end of the scale of estimates, and could be representative of just the separated plutonium at that time not including the amount used in the 2006 test and plutonium still inside the last load of fuel in the 5 MWe Reactor. It is feasible that the amount represents the two cores unloaded in 1994 and 2005 minus whatever was used in the 2006 test.⁽²⁰⁾

B. Facilities

B.1. The Fuel Fabrication Facility

The Fuel Fabrication Facility was built to provide fuel for the planned 200 MWe reactor, but its nominal throughput for most of the 1990's and 2000's has been about 100 metric tons of uranium metal fuel per year. The rate of production was set at this rate to provide enough fuel for the 5 MWe Reactor and for the initial loading of the 50 MWe that was planned. The input of the facility is natural uranium, in the form of yellow cake U_3O_8 . This material comes from the mining and conversion facilities located within North Korea. To produce 100 metric tons of fuel, it must process roughly 120 metric

tons of U_3O_8 .⁽¹¹⁾ The processes that occur in the Fuel Fabrication Facility are shown in Fig. 6.

The process begins with the dissolving of U_3O_8 in hot nitric acid. A 30% TBP mixture with kerosene is used in a solvent-extraction phase which is evaporated to produce uranyl nitrate. That uranyl nitrate solution is then heated to produce pure UO_3 . Pure hydrogen is added to produce UO_2 in a fluidized bed at around 600 C. The UO_2 is then moved to another fluidized bed where it is contacted by gaseous hydrofluoric acid at around 500 C to create UF_4 . The UF_4 is finely ground and reduced with magnesium chips and heated to 600-700 C from which a pure uranium metal ingot is produced. The uranium metal is alloyed with aluminum to create a 0.5% uranium-aluminum alloy. The alloy is extruded, heat treated, and machined into the final fuel rod dimensions and placed inside of a finned tube made of a magnesium-0.5% zirconium alloy. The fuel rod is finished by welding on an end cap before being placed in storage.⁽¹¹⁾

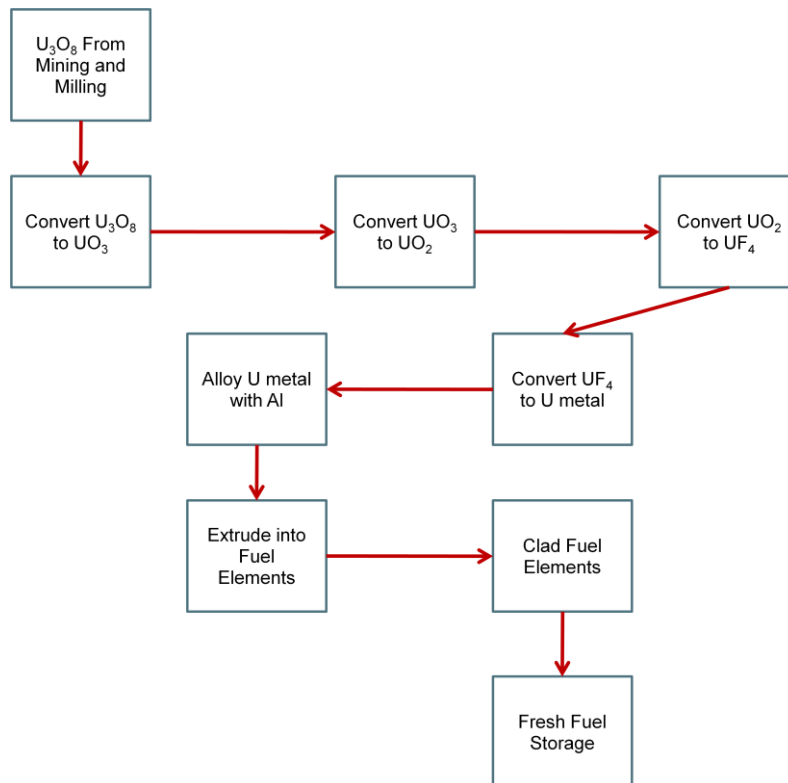


Fig. 6: Fuel Fabrication Facility Process Flowsheet

The 5 MWe Reactor requires 50 MT of fuel every two years. Assuming that the Fuel Fabrication Facility only runs at the capacity needed to re-fuel that reactor every two years, 25 MT of fuel rods needs to be produced every year which requires approximately 28 MT of U_3O_8 . If the proposed 200 MWe Reactor was finished, and assuming it was also refueled every two years, the Fuel Fabrication Facility would be required to make 1400 MT of fuel rods every two years. Therefore the capacity would be set at 700 MTU of fuel rods produced per year which would require 785 MT of U_3O_8 .

B.2. The 5 MWe Reactor

The 5 MWe reactor is a 20 MWth graphite moderated reactor cooled with CO_2 . Natural uranium metal is the fuel for this reactor, and the design is based off the first commercial electricity producing reactors at Calder Hall, built in the United Kingdom in 1956.⁽¹¹⁾ The final design is indigenous to North Korea, as are the materials used in construction.

The operating history of the reactor is unclear, but for the most part it is known that it operated with a high capacity factor from the second half of 1991 to April 1994 and again from February 2003 to July 2007. From January 1986, when the reactor first went critical, through the first half of 1991, the DPRK had some trouble operating the reactor and it did not have a high capacity factor.⁽¹¹⁾ The DPRK has admitted to burning 3 core loads of fuel, plus some replacement fuel for broken rods. However, there are sources that will dispute that claim, and assert that during the first run of the reactor, the core was at least partially refueled.⁽¹¹⁾ This would explain some of the discrepancies with the declaration that North Korea made in 1993. There also was an outage shown in satellite photographs of the cooling tower that was long enough to refuel the entire reactor in 1990. The DPRK has never admitted to shutting down and refueling the reactor during this time.⁽¹¹⁾

The design of the reactor has been published in journals, which allows for modeling in a transport-based fuel depletion code. For this research, the reactor will be modeled in TransLAT as a 2-d lattice.⁽²¹⁾ This code was chosen for the ease in which the geometry of the reactor can be input as well as the ability to simulate full core burnups with limited computing power. The TransLAT simulated model used is a 4x4 array of fuel rods surrounded by graphite with a control rod hole present in the middle of the array shown in Fig. 7. The array was used to generate average plutonium production values across the fuel pins which can be translated into total plutonium production for the core. The model was also used to generate average cross sections for use in a two group, two region diffusion problem to solve for the power profile.

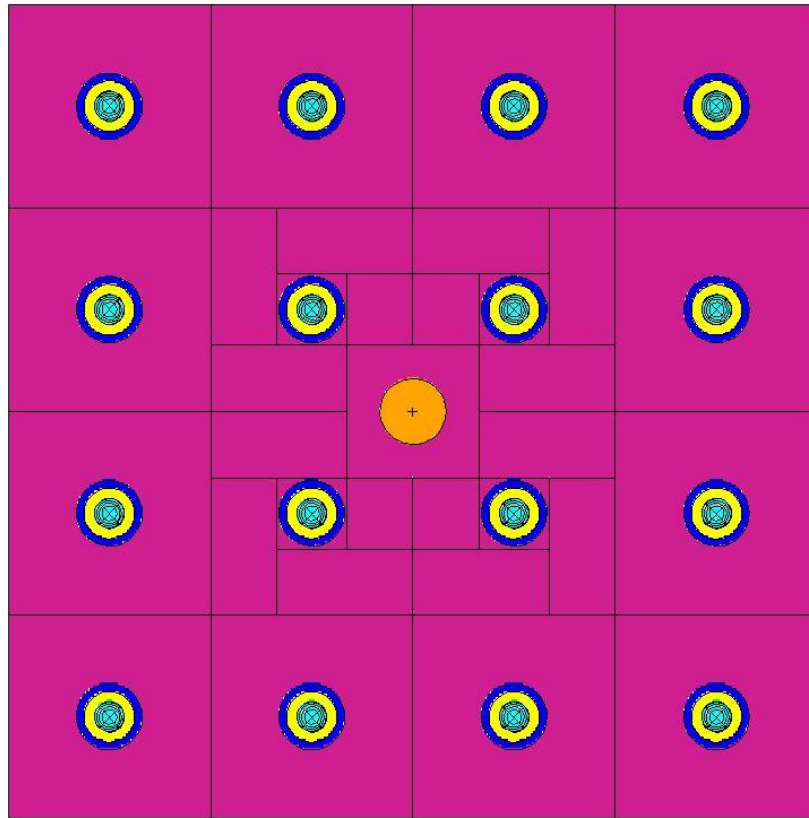


Fig. 7: TransLAT Model of the 5 MWe Reactor

The model used is a 4x4 array of fuel rods surrounded by graphite with a control rod hole present in the middle of the array. The array was used to generate average

plutonium buildup values across the fuel pins which can be translated into total plutonium buildup for the core. The model was also used to generate average cross sections for use in a two group, two region diffusion problem to solve for the power profile. TransLAT is a 3-d deterministic lattice physics burnup software that is used to evaluate the performance and characteristics of different reactors.⁽²¹⁾

The 5 MWe reactor has an abnormal axial power profile due to the way in which it is run. When it is initially started, it is run with control rods inserted about one third the way into the core. This causes a non-uniform axial power profile which affects the burnup of the fuel in the core. Power peaks much more severely than the normally assumed cosine shape of a standard power profile. As a result, some fuel rods received an abnormally high burnup close to 1,370 MWth-day/metric ton of uranium while others received only 64 MWth-day/metric ton of uranium during the initial core loading.⁽¹¹⁾ Because of this abnormal axial power profile, the total ^{240}Pu content from the reactor will be slightly higher than it would have been at the same average burnup with a cosine shaped power profile. To calculate this new profile, a simple two region one group diffusion calculation is made using averaged cross sections of all the materials in each region.⁽²²⁾

Averaged cross sections were generated in TransLAT for the 4x4 assembly with control rod material in the center hole and with CO_2 as well. These cross sections were used in the two region, two group diffusion system of equations and solved numerically using a software package named Maple.⁽²³⁾ The Maple worksheet can be found in Appendix B. The calculated power profile of the reactor can be found in Fig. 8.

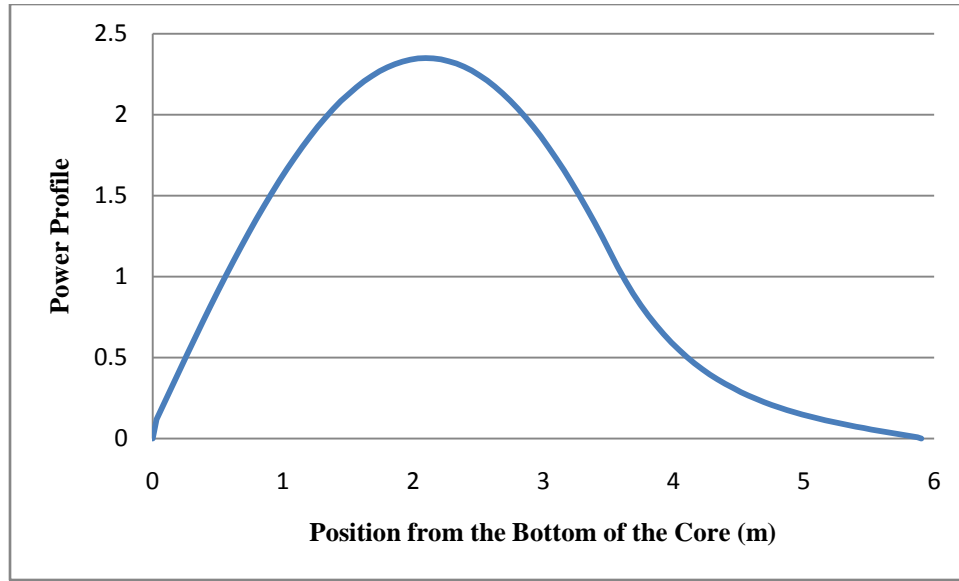


Fig. 8: Power Profile

The power profile is significantly depressed in the region with the control rod. This forces a higher power in the lower part of the core, which is where the materials of the reactor will be most stressed. The power profile can be used to determine the centerline temperature of the fuel, as well as the temperature distribution from fuel centerline through the bulk temperature of the coolant.⁽²²⁾ To obtain those temperatures, the temperature of the coolant is first calculated using:

$$T_{\text{Bulk}}(z) = T_{\text{in}} + \frac{q_{\text{Max}}''' \cdot 2.32 \cdot A_{\text{channel}} \cdot H}{\dot{m} \cdot c_{p, \text{CO}_2}} \cdot \int_0^z P(z) dz \quad (1)$$

where T_{in} is the inlet temperature of the coolant, q_{Max}''' is the maximum volumetric heat generation rate, A_{channel} is the area of the coolant flow channel, H is the active height of the core, \dot{m} is the mass flow rate of the coolant, c_{p, CO_2} is the specific heat capacity of the CO_2 coolant, and $P(z)$ is the power profile. The 2.32 constant reflects the radial power profile, it is the peak to average ratio of the zero order Bessel function that governs the radial power profile. Eq. 1 is then used to determine the temperature of the inner cladding surface:

$$T_{\text{CI}}(z) = T_{\text{Bulk}}(z) + \frac{2\pi \cdot q_{\text{Max}}''' \cdot r_{\text{CI}}^2}{2 \cdot k_{\text{clad}}} P(z) \cdot \ln \frac{r_{\text{CO}}}{r_{\text{CI}}} + \frac{2\pi \cdot q_{\text{Max}}''' \cdot P(z)}{h \cdot r_{\text{CO}}} \quad (2)$$

where r_{CI} is the inside cladding radius, k_{Clad} is the conductivity of the cladding, $P(z)$ is the power profile at that location, r_{CO} is the outer radius of the cladding, and h is the enthalpy of the coolant. This point is the highest temperature in the clad. Due to the cladding being a magnesium alloy, the working temperature is around 650 °C; above that temperature, the magnesium will lose structural strength and be unable to support the mass of the fuel. In order to guarantee the integrity of the reactor and the structural stability of the fuel, the cladding temperature will be limited to 600 °C. Finally, the maximum temperature of the fuel is calculated by:

$$T_{CL}(z) = T_{CI}(z) + \frac{q'''_{Max} \cdot r_{CI}^2}{k_{Fuel}} \quad (3)$$

where k_{Fuel} is the conductivity of the fuel and $T_{CL}(z)$ is the temperature of the centerline of the fuel. The melting temperature of uranium metal is 1132.2 °C, and the limit will be set at 1000 °C.

Using the previous equations with the already established power profile, we can generate the temperature profile in the hottest channel in the reactor. As can be seen in Fig. 9, the limiting factor is the temperature of the cladding. At the rated power of 20 MWth, the cladding reaches a maximum temperature of around 590 °C, which is close the 600 °C limit. Therefore, it can be safely assumed that the power of the reactor is limited to the stated maximum power.

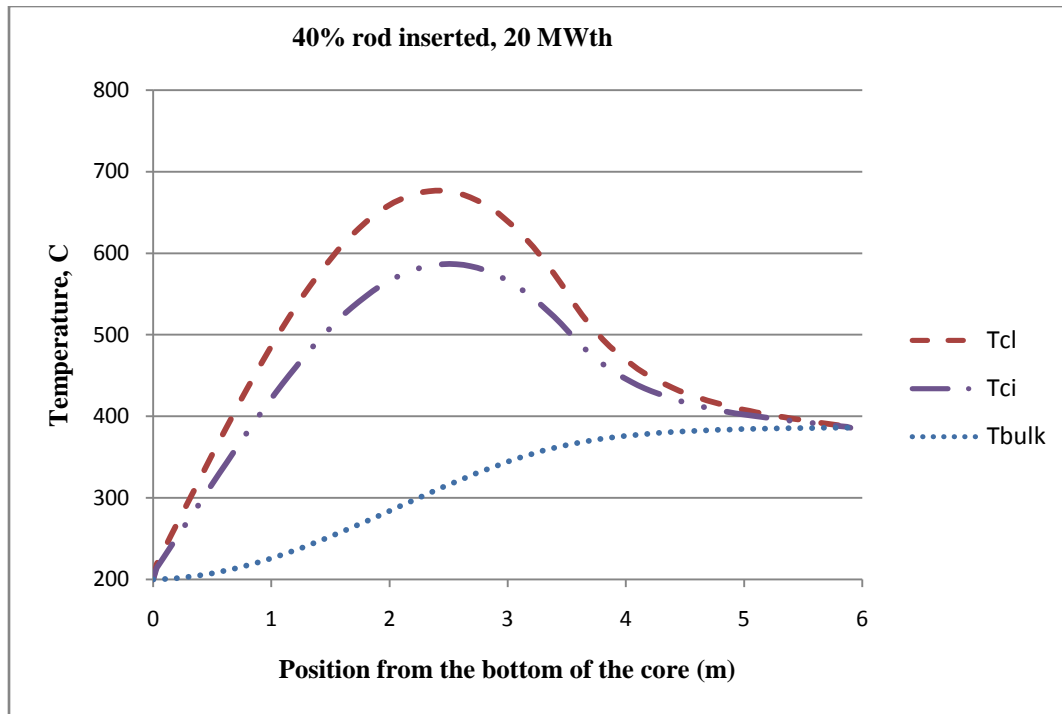


Fig. 9: Temperature Profile of the 5 MWe

The TransLAT model will be used to estimate the plutonium production of the reactor. TransLAT will give concentrations of isotopes in the spent fuel, which can be converted into an amount of plutonium per core load. For this test case, several different burnup steps will be simulated. For the last two completed core loads, the fuel has been burned around 2 years before being replaced. Using that time step, it is possible to calculate an average burnup of the core for that amount of time. It will be assumed that the reactor is run for approximately 300 days per year or 600 days per core. The amount of fuel initially loaded into the core is estimated to be about 50 MT of uranium metal. The full thermal power of the reactor will be assumed to be a constant 20 MWth. Burnup is calculated by:

$$\text{Burnup} = \frac{\text{Power (MWth)} \cdot \# \text{ of Days at Power}}{\text{Mass of Fuel in MT}} \quad (4)$$

reactor power is set at 20 MW(th) and the mass of uranium in the fuel is 50 MT.

From that equation, the average burnup of the fuel from the February 2003 through April 2005 burn can be estimated to be about 240 MW days per metric ton by assuming

300 days of operation per year over a two year time span. TransLAT will be used to simulate several potential burnup steps, from fresh fuel to 300 MW days per metric ton. Fig. 10 shows those estimated values.

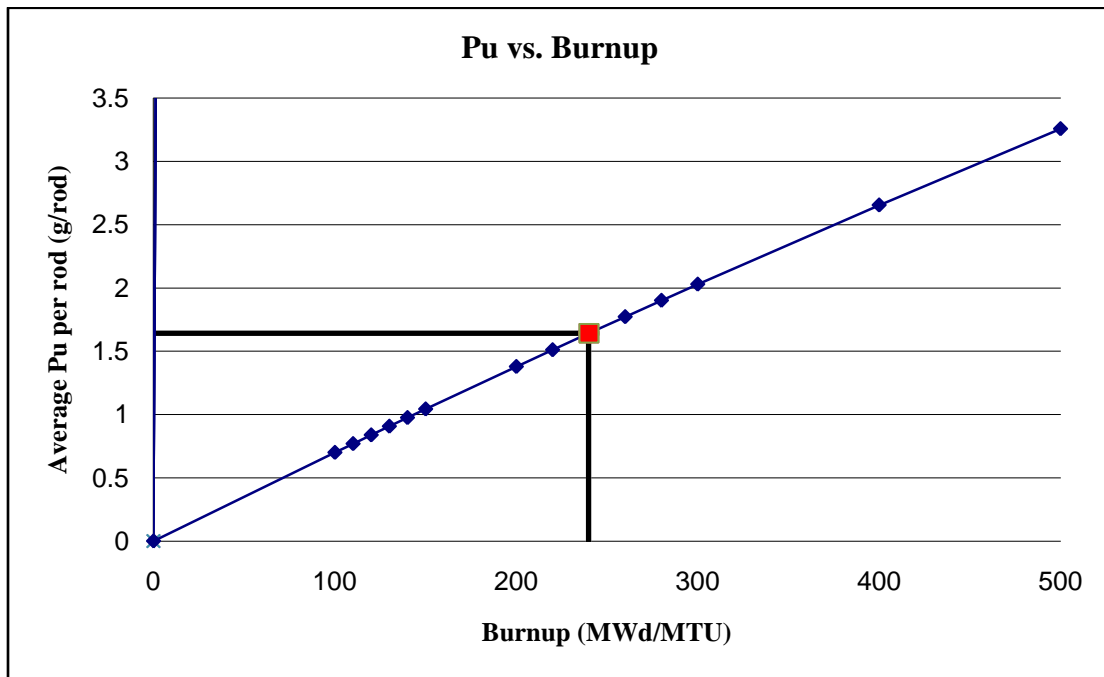


Fig. 10: Plutonium Production in the 5 MWe Reactor

The values that TransLAT has estimated are on the upper end of the ISIS estimates of 6-7 kg per year of plutonium. The estimates may be a little high compared the actual production, but for the purposes of this test case it is acceptably conservative. The ^{239}Pu content of the spent fuel from this reactor is also seen to be very high at different burnups in Fig. 11.

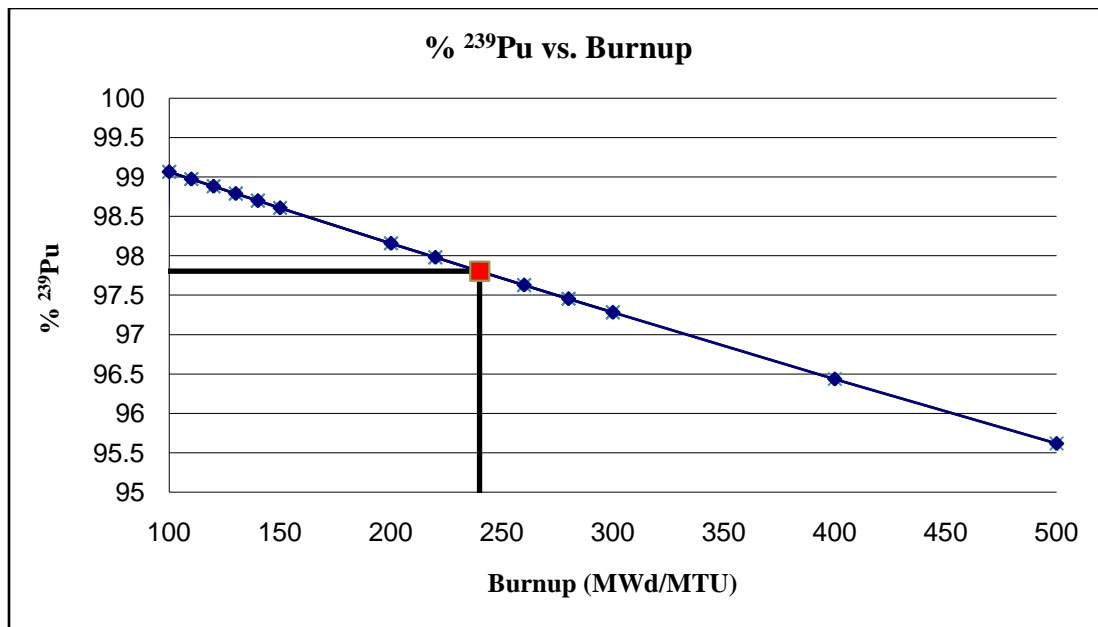


Fig. 11: ^{239}Pu Content of the 5 MWe Spent Fuel

B.3. The Radiochemical Laboratory

The Radiochemical Laboratory is the key facility to safeguard in this fuel cycle. If the material passing through this facility can be adequately safeguarded, the pathway for the state to generate separated plutonium is severely hampered. This facility is an industrial scale reprocessing facility capable of reprocessing 220-250 tons of spent nuclear fuel per year, which is enough for both the 5 MWe and the proposed 50 MWe reactors. There are two lines that are completely independent. The facility uses the ‘chop-leach’ PUREX process with chemical decladding.⁽¹¹⁾ A general flow sheet for the processes of the Radiochemical Laboratory is shown in Fig. 12.

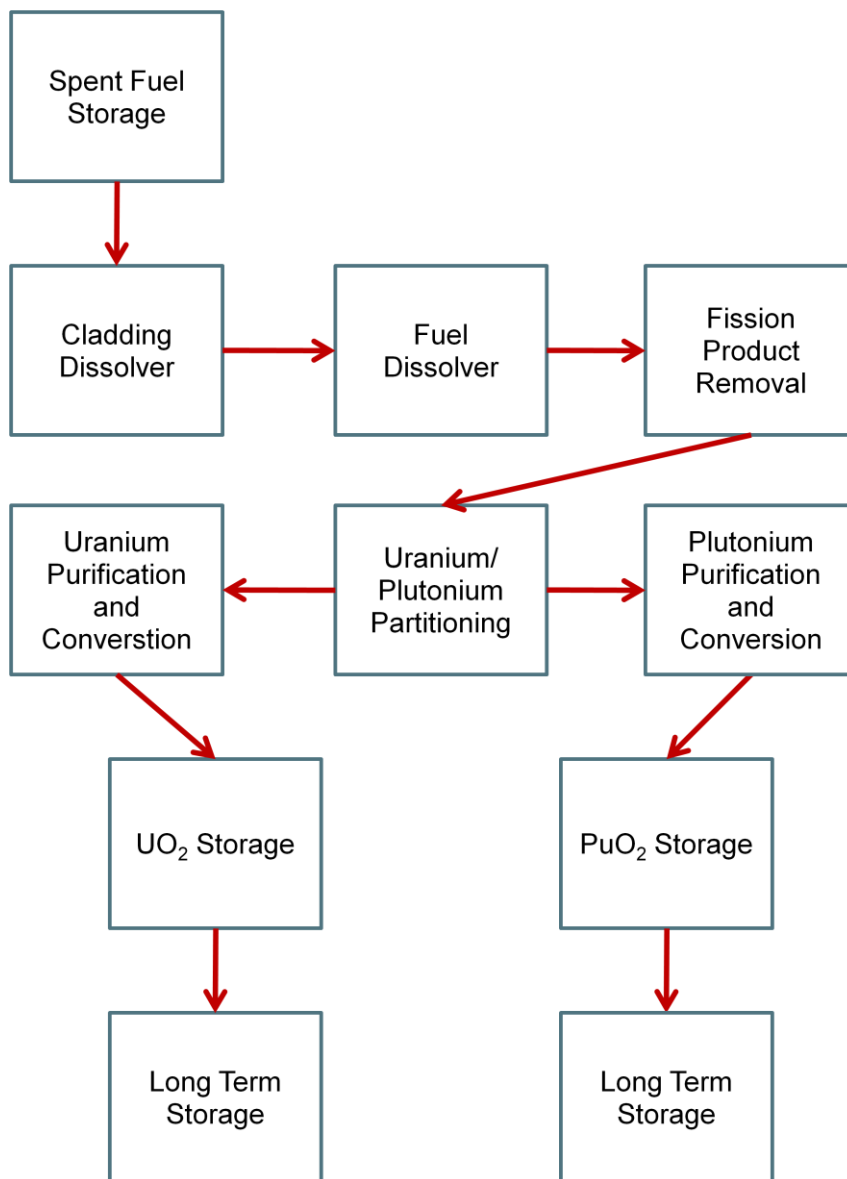


Fig. 12: Process Flowsheet for Radiochemical Laboratory

The PUREX process used at the Radiochemical Laboratory is very similar to that of commercial facilities with very few changes.⁽²⁴⁾ The fuel is processed in batches of 20 rods, and each rod has structural pieces, like end-sections and grappling tabs removed. From there, a hot, dilute nitric acid mixture is added to dissolve away the magnesium-zirconium cladding. Each batch is transferred to another dissolver and broken up into smaller, 5 rod batches. Hot, concentrated nitric acid is added to each of

the smaller batches to dissolve the fuel rods. All of the volatile and some of the semi-volatile fission products are removed from the fuel and released as off gas at this time. There is an accountability tank that is used to sample and analyze the spent fuel solution that now contains nitrates of uranium, plutonium, and nonvolatile fission products.⁽²⁵⁾

From the accountability tank, the solution is pumped into several mixer-settlers and mixed with a 30% TBP-kerosene in the first decontamination cycle. There are about 30 mixer-settlers per line in this decontamination cycle, each with an 80 liter capacity. Three aqueous streams are pumped into the bank: one of the fuel dissolver solution, a uranium stripping solution, and a plutonium stripping solution. Three aqueous streams are also removed: one of fission products, one of uranium, and one of plutonium. The organic stream strips all but about 2% of the uranium and plutonium together and leaves the fission products in the aqueous stream. The uranium and plutonium rich organic stream is then stripped of uranium using dilute nitric acid. The plutonium is then stripped using an aqueous nitrate solution of hydroxylamine and hydrazine. The organic stream is available for re-use after this stripping step.⁽²⁵⁾

The plutonium stream is then purified through another round of solvent extraction. In a commercial plant, this would also be done on the uranium stream, however the DPRK has never implemented a uranium purification stream. They classify the aqueous uranium stream as waste.⁽¹¹⁾

To estimate the material flows through this facility, there are a few small calculations that need to be made. First, it is known that a full core load of 8000 rods were reprocessed in the first six months of 2003. Knowing that the facility is run in batch mode with two lines present using 20 rods per batch, it is fairly easy to see that roughly 2.5 batches are completed every day on average. Using the TransLAT simulation to determine a nominal plutonium output for a core, the average amount of plutonium per fuel rod and subsequently per batch can be calculated. About 100g of plutonium enters the facility on average per day.

CHAPTER III

PROPOSED SAFEGUARDS

In the previous chapter, the Yongbyon fuel cycle was fully mapped out and the material throughput of the facilities was stated. That information will be used to design a material accountancy program around the fuel cycle to safeguard the material from being used in a weapons program. However, there is one facility that is already under a safeguards agreement and will not be included in this project. The IRT reactor is still under an INFCIRC-66 type agreement, and there is no need to change that. In its current configuration, the IRT is capable of producing about 0.4 kg of plutonium per year by irradiating natural uranium targets.⁽²⁶⁾ As will be seen later, this amount is much lower than the anticipated uncertainty associated with the plutonium measurements inside the Radiochemical Laboratory. Therefore the plutonium production of the IRT can be ignored in this safeguards arrangement. The fuel cycle will also be modeled with some slight modifications to the way in which material flows through it due to unfinished construction and the need to safeguard uranium in the spent fuel. The modeled fuel cycle is shown in Fig. 13. The 50 MWe and the 200 MWe Reactors were never finished and the Radiochemical Laboratory does not currently recover uranium from spent fuel. That uranium must also be safeguarded. The modeled fuel cycle will include a uranium recovery stage in the Radiochemical Laboratory.

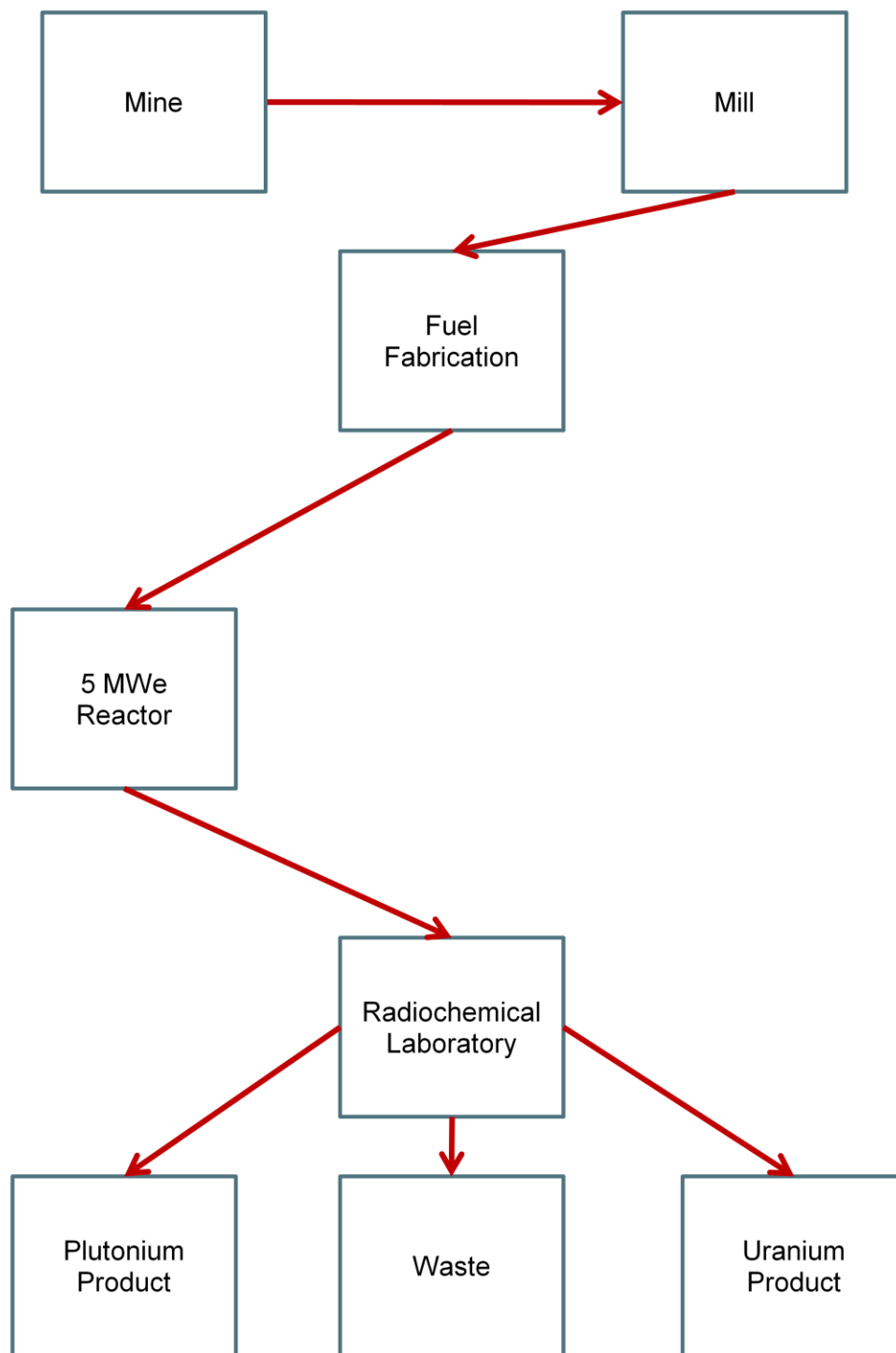


Fig. 13: Modeled Fuel Cycle

A. The Fuel Fabrication Facility

Each process in the fuel fabrication plant changes the chemical form of the uranium, not the isotopic ratio of ^{235}U to ^{238}U , which simplifies the safeguards process. The material throughput and processes of this facility have been researched; therefore the analysis will begin by establishing the safeguards goals for this facility.

Because this is a fuel cycle based off of natural uranium, the timeliness goal mandated by the IAEA for detection of a significant quantity is 75kg of ^{235}U within 3 months. The facility is theorized to be capable of producing enough fuel to adequately fuel the planned but never completed 200 MWe reactor.⁽¹¹⁾ The proposed system will be tested for two different capacities, one capacity based off of the needs of the 5 MWe reactor and another based off of the proposed needs for the 200 MWe reactor. Only one MBA will be used in the proposed safeguards system for the fuel fabrication plant, and it will cover the entire process from when the U_3O_8 enters the facility to the finished fuel rods of the alloyed uranium metal exit the facility for fresh fuel storage. The same MBA will be used for both capacities, and the same type of detectors will be used to measure input and output. Also, there will be no simulated losses of material inside the facility.

In order to adequately safeguard this facility, an analysis of the potential pathways that an inside entity would use to divert material must occur. This facility has only indirect-use material, so its contents have less stringent requirements than a facility that actually enriched uranium or handles plutonium. Natural uranium in any of the chemical forms found inside the facility is not a direct use material. Taking material from any point in the process from U_3O_8 to U metal will never give the proliferator a suitable bomb making material. Furthermore, the most common material used in enrichment facilities, UF_6 , is not produced or stored at this facility. The most likely material diverted from this facility would be either the U_3O_8 as it is brought into the facility, or the uranium metal fuel rods as they leave the facility. The nuclear material accountancy program at the fuel fabrication facility must accomplish three goals: it must verify that it receives all of the U_3O_8 from the mining and milling operation, it must

guarantee that no material is lost during the process from U_3O_8 to uranium metal, and it must account for each fuel rod that is shipped off to the reactor.

To accomplish those goals, an MBA will be placed around the facility covering from when the U_3O_8 enters the facility through the finished fuel rod complete with cladding exiting the facility and placed in fresh fuel storage. The amount of ^{235}U needs to be measured on the way in and on the way out. To find the amount of ^{235}U in the U_3O_8 , we need to know the enrichment of the material and the mass of the U_3O_8 coming into the MBA. To calculate the amount of ^{235}U in the U_3O_8 the following equation is used:

$$\text{Mass of } ^{235}U = \text{Mass of } U_3O_8 \cdot \frac{\text{Mass of uranium}}{\text{Mass of } U_3O_8} \cdot ^{235}U \text{ Enrichment} \quad (5)$$

where the mass of U_3O_8 and the ^{235}U enrichment is measured and the amount of uranium in U_3O_8 is a constant based off the molar weights of uranium and oxygen. This equation is made up of constants and measured values. For the incoming U_3O_8 , the mass fraction of uranium in the U_3O_8 will always be the same for a given enrichment. In this case, with natural uranium and oxygen, the atomic masses from any periodic table can be used to calculate this quantity. It is assumed to have a negligible uncertainty with regards to the other two measurements, so it will be ignored in the uncertainty analysis. The other two parts of the equation will be measured. Normally, U_3O_8 is shipped using 30 or 55 gallon drums. These drums are heavy enough that to measure their mass, it would be beneficial to use a Load Cell Based Weighing system (LCBS). For the enrichment measurement, there are a few options, and the choice is dependent upon the capacity of the facility. There are many different systems that can measure enrichment, both by destructive analysis (DA) and non-destructive analysis (NDA). NDA is easier to perform, cheaper, and faster than DA, however the results have higher uncertainties. For the DPRK program, the throughput of this facility allows the higher uncertainty of NDA to be acceptable. A Portable Multi-Channel Analyzer coupled with a NaI detector will have adequate uncertainties for this facility when it is producing fuel for the 5 or 50 MWe reactors, but the uncertainties begin to be unacceptable for the needs of the proposed 200 MWe reactor.

While the NaI detector is certainly capable of determining the enrichment of the uranium, its higher uncertainty forces the Material Balance Period to 14 days. If the fuel fabrication facility ever begins processing 700 MT/yr of material, a better measurement with lower uncertainties will most likely be needed at both the input and output of the facility. One way to do that would be to use a High Purity Germanium (HPGe) detector in place of the NaI detector. NaI detectors have a random and systematic uncertainty of 5%, while HPGe detectors have a random uncertainty of 3% and systematic uncertainty of 2%.⁽²⁷⁾

As can be seen in Fig. 14, the fuel fabrication MBA is fairly easy to set up. The overall throughput is relatively small compared to other facilities worldwide when it is just producing fuel for the 5 MWe Reactor. There is no enrichment of ^{235}U , and there is no presence of plutonium in this facility. This is easily accomplished by the following.

As the uranium enters the facility, it will be weighed and have its enrichment measured. Both of these measurements will have uncertainties with them. Those uncertainties will be taken from the ITV-2000 which is a document produced by the IAEA that defines the expected random and systematic uncertainties for many detectors commonly used in nuclear safeguard systems. For the LCBS and the EBAL weighing devices, the random and systematic uncertainty is the same at 0.5%.

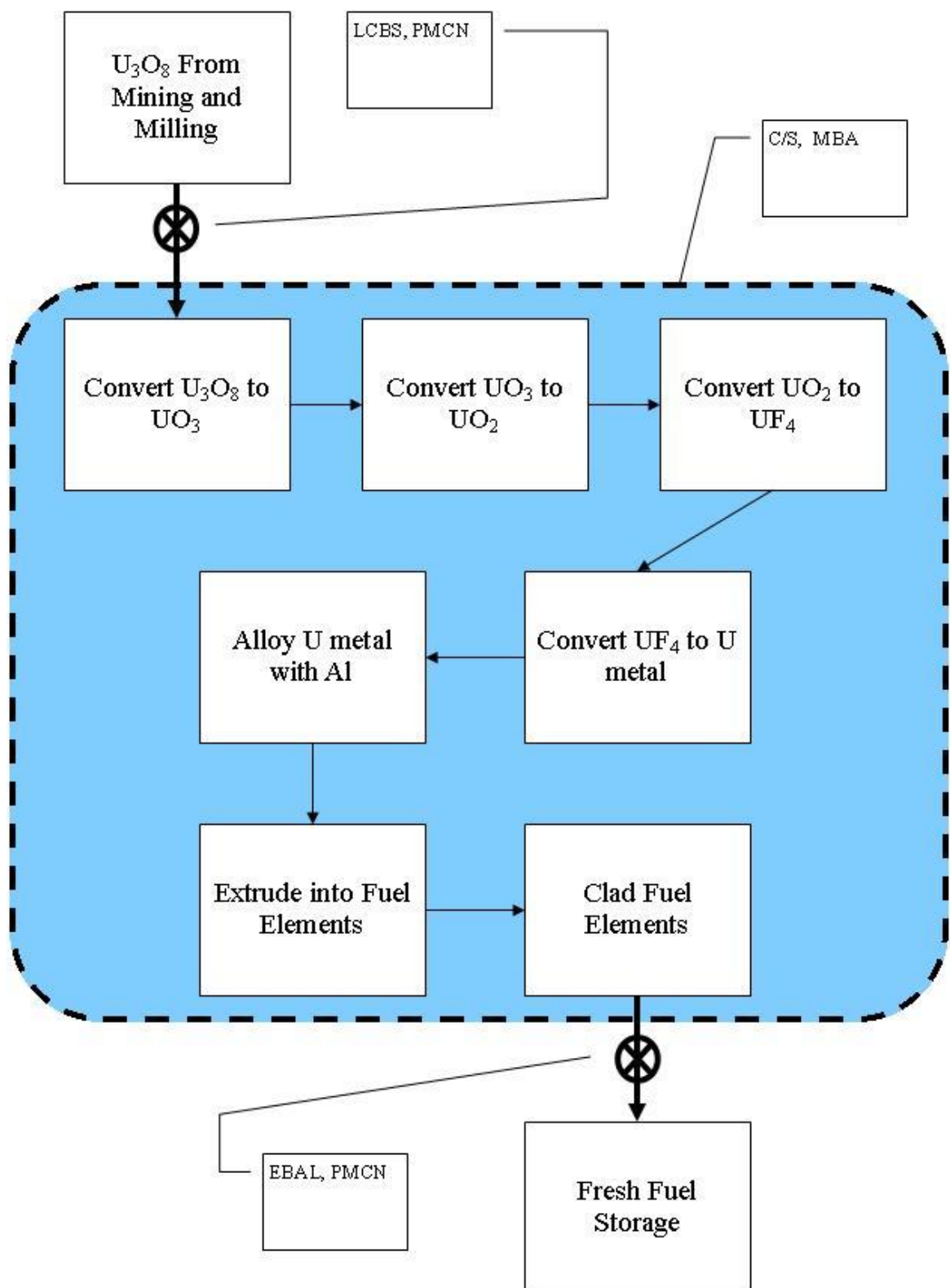


Fig. 14: Proposed Fuel Fabrication MBA Set Up

This facility will be analyzed for 2 different scenarios, the first at its hypothesized full capacity which is enough to fuel the planned 200 MWe reactor and the second with the capacity set for only the needs of the 5 MWe Reactor. It will be assumed that both reactors run on a similar cycle as the 5 MWe Reactor did for the 4 years it was restarted, 2003-2007. With that assumption, the two capacities that will be simulated are 700 MT and 25 MT of fuel rods that will need to be made per year. Table 2 shows the 700 MT/yr analysis.

Table 2: Fuel Fabrication Uncertainty Analysis at 700 MT/yr

Input			Output		
U ₃ O ₈ Input	2740	kg/day	U metal	2320	kg/day
²³⁵ U Input	16.7	kg/day	²³⁵ U Output	16.7	kg/day
MUF input	16.5	kg	MUF output	16.5	kg
Sigma MUF	23.4	kg	Capacity	700	MT/yr
IAEA Target	25	kg	MBP	14	days

As can be seen in the table, a very short MBP is required for this capacity. There is a great quantity of material being processed in the facility per day which drives up the uncertainty of any measurement. To increase the MBP at this capacity, a better measurement is needed. If the NaI detectors are replaced with HPGe detectors, the MBP can be increased to over 140 days, which is a much better figure. Longer MBP's allows for less shutdown time to flush out the processes in a facility which saves the operator money.

Table 3 shows the uncertainty analysis of the Fuel Fabrication Facility at the capacity of 25 MT/yr. For this capacity, the MBP can be much longer with the NaI detectors. No MBP can exceed the timeliness goal for detection of diversion, therefore the limit on the MBP for this case is the timeliness goal.

Table 3: Uncertainty Analysis of the Fuel Fabrication Facility at 25 MT/yr

Input			Output		
U ₃ O ₈ Input	97.8	kg/day	U metal	82.9	kg/day
²³⁵ U Input	0.597	kg/day	²³⁵ U Output	0.597	kg/day
MUF input	15.4	kg	MUF output	15.4	kg
Sigma MUF	21.8	kg	Capacity	25	MT/yr
IAEA Target	25	kg	MBP	365	days

The MBA that has been set up in both cases will accomplish the two goals of safeguards. The material is measured as it enters the facility, this value can be checked against the information sent from the mill and should match. If there is any material diverted from one of the chemical processes inside the facility, less material would be measured in the finished fuel rods. The addition of a substitute material to make up the weight of any diverted uranium would also be detected by the gamma spectroscopy equipment. The finished fuel rods can be item accounted as they leave the facility, and this number should match the number of rods created.

B. The 5 MWe Reactor

The 5 MWe Reactor is the easiest facility to safeguard in the entire fuel cycle. Self contained fuel rods come in, and the amount of uranium and its enrichment can be verified easily, and self contained spent fuel rods leave. Item accounting will work for this facility, matching the number of rods in and the number of rods out is simple work. Fig. 15 shows the MBA set up for the 5 MWe reactor.

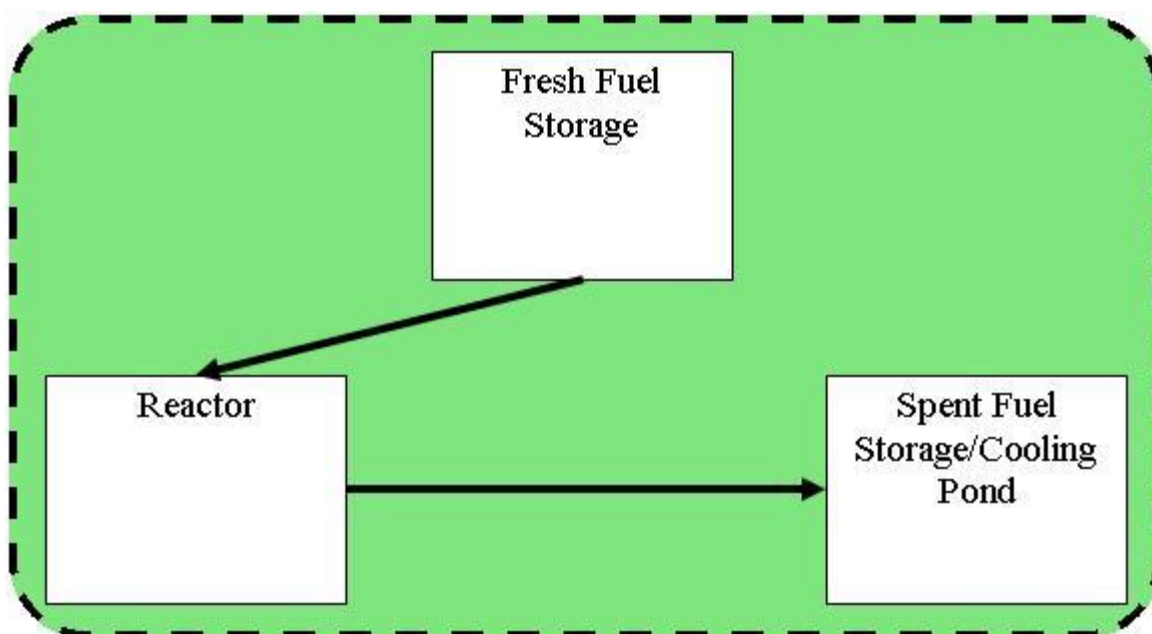


Fig. 15: Proposed 5 MWe Reactor MBA Set Up

The most common diversion pathway for a reactor involves removing a single fuel rod from an assembly. This fuel rod can be from either fresh fuel or spent fuel. For this reactor, that pathway will not be available because the fuel is loaded as individual rods, not assemblies. There is still the potential for a single rod to be diverted, but that diversion would be accounted for when the number of fuel rods leaving the reactor did not match the number of fuel rods entering the reactor.

The reactor must be more carefully analyzed to fully guarantee total material accountancy. The reactor could be run at a higher than stated power level, which would cause an increase in the amount of plutonium that is created for the same amount of irradiation time. Also, if there is a covert supply of fuel, it is possible to achieve the same burnup on the fuel in a shorter amount of time, which would allow the reactor to have the fuel replaced sometime during a normal cycle to allow for covert plutonium production with undeclared fuel. In general, the reactor needs to be analyzed for both what is stated by the country as normal operation as well as the maximum capacity in terms of plutonium production. In this case, as Chapter II shows, this reactor is operating at very close to the thermodynamic limits of the materials in the reactor.

Running the reactor at a higher power would have a higher probability of causing some of the reactor materials to fail. Therefore, the operating history of the reactor should be verified through measuring the burnup of the spent fuel, but the day to day operation of the reactor does not need to be monitored. The fuel loading machine should also be locked down and sealed during reactor operation.

For the accountancy system at the 5 MWe reactor, it is all item accounting. Fuel comes in as discrete rods, which will be counted as they enter the fresh fuel storage. Each rod will be counted also as it is loaded into the reactor, and as it leaves the reactor. To guard against switching fuel rods for rods without fissile material, each rod will be passed through an ion chamber as it is pulled from the reactor by the fuel loading machine. Spent fuel rods will have a fairly high activity, while any other material will have a different amount of activity.

C. The Radiochemical Laboratory

The last step in the fuel cycle to be analyzed is the Radiochemical Laboratory. This facility is the sole location in the entire fuel cycle that at some point has separated fissile material in a form that is readily used for weapons purposes. The types of processes that occur in the facility were explained in Chapter II, and the amount of material that this facility can process was also explained. Unlike the Fuel Fabrication Facility, there is a direct use material present in this facility. Plutonium and plutonium bearing materials are a much better target for a proliferator because once the plutonium has been chemically separated it is weapons usable. There are many barriers that naturally slow down a proliferator, however. As spent fuel enters the facility, it is highly radioactive and not easy to handle. That is a barrier that a proliferator would have to overcome. Each step in the process represents a different challenge for a proliferator, but the same problem for a safeguards system. Therefore, one MBA can be used from when the spent fuel is first loaded to be dissolved through the storage of the uranium and plutonium products.

The diversion pathways of the Radiochemical Laboratory are classic for any PUREX process. For this project four pathways will be analyzed: diverting a spent fuel rod from the spent fuel storage, diverting dissolved fuel, diverting the dissolved fuel after it has been cleansed of fission products, and diverting PuO_2 from storage. These four pathways are the classic diversion scenarios for a PUREX facility, and they test the integrity of the proposed accountancy program adequately.

Normally, there are no measurements of plutonium content until after the fission products are removed from the spent fuel solution. The reasons behind that decision are two-fold, first the plutonium signal is drowned out by other isotopes in the spent fuel, like curium, and second the uncertainties associated with such measurements are very high, on the order of 25%. However, because the total amount of plutonium is small in this case, and the timeliness goal is relatively large compared to the amount of plutonium that is inside the facility at any given time, that amount of uncertainty is actually acceptable for a facility of this size. For this facility, a full accountancy program will work and the needs for containment and surveillance are lessened because of it. The bulk of the facility can be placed inside one MBA that encompasses everything from the cladding dissolution through the purification of the uranium and plutonium streams. For the purposes of this analysis, it will be assumed that the Radiochemical Laboratory will have the uranium purification step added to its process line. This assumption is made to lower the total amount of material that is considered waste that would contain large amounts of uranium which must also be safeguarded. There are two exits from the main MBA, one from the purified uranium stream and one from the purified plutonium stream. Fig. 16 shows the MBA set up for the Radiochemical Laboratory.

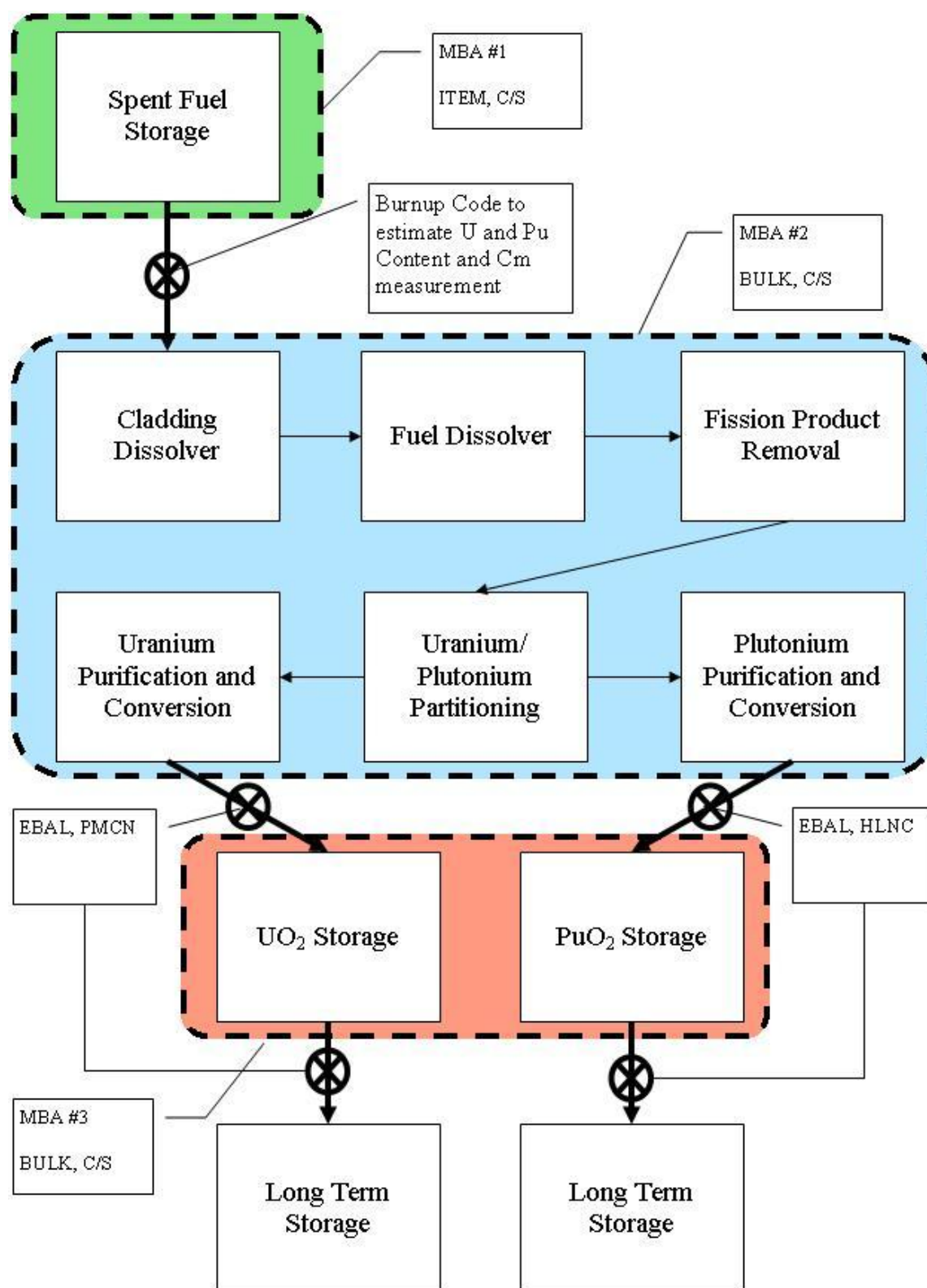


Fig. 16: MBA Set Up for the Radiochemical Laboratory

To estimate the material flows through this facility, there are a few small calculations that need to be made. First, it is known that a full core load of 8000 rods were reprocessed in the first six months of 2003. Knowing that the facility is run in batch mode with two lines present using 20 rods per batch, it is fairly easy to see that roughly 2.5 batches are completed every day on average. Using the TransLAT simulation to determine a nominal plutonium output for a core, the average amount of plutonium per fuel rod and subsequently per batch can be calculated. About 100g of plutonium enters the facility on average per day. The average amount of plutonium going through the MBA per MBP is then found by estimating an appropriate MBP and multiplying the average amount of plutonium going through per day by the days in the MBP.

At this point, the detection methods need to be decided. This facility will be analyzed in two cycles, a plutonium cycle and a uranium cycle. The material entering the facility will be measured using a combination of a computer code simulation of the core combined with a measurement of the activity of the spent fuel to estimate the burnup. As can be seen in Fig. 10 from the previous chapter, plutonium will build up almost linearly with burnup, which means an approximation of the plutonium content of the fuel can be made. As previously mentioned, the uncertainty of this measurement is very high, but in this case acceptable due to the small amounts of material that are passing through the facility per day. On the other end of the process, there is a refined product of pure PuO_2 in powder form, which makes this a bulk material. To quantify the amount of plutonium at that end the High Level Neutron Coincidence Counter (HLNC) will be used. The HLNC is a device that passively measures the amount of plutonium in anything from sub gram to 10 kg size quantities. It also has a fairly low uncertainty associated with its measurement. Now, with the measurement system determined, the actual simulated accountancy can begin. For this purpose, each chemical process will be considered to have some loss of material associated with it. These losses are meant to model a real facility, and are based off of published data on the Hanford plutonium extraction efficiency from the 1960's.⁽²⁸⁾ For this simulation, it is assumed that about

0.7% of the plutonium is lost in the chemical processes from beginning to end, which is slightly higher than the published results. The estimated loss is higher than the published data for a few reasons. First, the higher losses will inflate the amount of MUF calculated, which will cause a more conservative MBP than assuming no losses have occurred. Second, the amount of loss is probably higher for the DPRK due to a less developed chemical industry and expertise.

The following table shows the breakdown of plutonium over various burnups.

Table 4: Plutonium Amount used in Simulation

Burnup of 240 MWd/MT	
Simulated Total Pu Per Core (g)	13100
Average Pu per rod (g)	1.64
Average Pu per batch (g)	32.8
Average Pu per day (g)	82.1
Average Pu per MBP (g)	2460

In Table 4, an MBP of 30 days was used to calculate the plutonium entering the facility per MBP. As in the Fuel Fabrication Facility, there can only be one MBP per facility and it is limited by the length of the shortest timeliness goal applicable to the facility based on the types of material inside the facility. Therefore, the maximum MBP for the Radiochemical Laboratory was set at 30 days. The TransLAT results give a reasonable amount of plutonium produced per core load every two years, on the high side of the estimates released from ISIS. Simple arithmetic was used to calculate the other values in the table knowing that each core is composed of 8000 rods, each batch is made up of 20 rods, and about 2.5 batches are completed each day. Table 5 shows the simulated losses in the chemical process.

Table 5: Waste Plutonium Estimation

MBA #2 Pu	Pu (g)	Waste Pu (g)
Pu after Decladding (g)	2440	19.7
Pu after Dissolving (g)	2440	3.66
Pu after FP Removal (g)	2430	3.66
Pu after Purification (g)	2430	2.43

These values were estimated by assuming a flat 0.8% loss of material in the chemical decladding phase, 0.15% loss of material in both the fuel dissolving and fission product removal stages, and a 0.1% loss in the purification step. A total loss of around 0.7% of the material is achieved this way, which is in line with other PUREX facilities around the world.

Now that the amount of simulated material is known at each step, it is possible to evaluate the proposed system of detectors to calculate the MBP that will be required to achieve the timeliness goal. The mass entering the second MBA, the mass that will be measured using the burnup correlation model, will be called M1 for this simulation. The mass leaving the second MBA after the purification step, the mass being measured by the HLNC will be called M2. From the Passive Nondestructive Assay of Nuclear Materials, the uncertainty of a burnup correlation to plutonium production has a systematic uncertainty of 25% and a random uncertainty of 3.9%.⁽¹⁴⁾ Combining the two figures the uncertainty the M1 measurement is calculated to be 25.3%. From the ITV-2000 publication, the HLNC has a random uncertainty of 1% and a systematic uncertainty of 0.5% for a combined uncertainty of the M2 measurement of 1.12%.⁽²⁷⁾ Using the 30 day MBP of Tables 4 and 5, the following results in Table 6 can be found.

Table 6: Uncertainty Analysis on Pu Measurements

M1 Uncertainty (g)	623
M2 Uncertainty (g)	27.2
MUF Uncertainty (g)	623
IAEA Target (g)	2670
MBP = 30 days	

The 30 day MBP works and the 3 sigma value is below the 8 kg significant quantity in the timeliness goal. Having an MBP this long is very desirable for many reasons. Knowing that with these relatively inexpensive detector systems a material accountancy program can be sustained means that there is some leeway in the decision on which detector to use. Also, this MBP allows the IAEA to have fewer inspections which can be costly. Once the plutonium cycle has been effectively safeguarded, there is one other aspect to consider, the uranium cycle.

The current layout of the Radiochemical Laboratory does not have any steps designed to extract uranium from the spent fuel solution. In its present operation, that uranium is considered waste. However, it does not have to be wasted, and it could be sold to be re-enriched or for some other use. It is slightly depleted uranium, and should also be placed under safeguards. For the purposes of this simulation, it will be assumed that the last part of the cycle has been completed and that uranium will be separated and purified for either long term storage or for use later. The same MBA's will be used and some of the same equipment will be used to measure the ^{235}U inside the uranium. Again, a burnup simulation will be done to estimate the amount of uranium and what its enrichment is for a given burnup. The same uncertainties still apply.

For this part of the cycle, only the amount of ^{235}U is safeguarded, and the IAEA's timeliness goals are 75kg of ^{235}U within 3 months. Therefore, only ^{235}U will be tracked in this simulation. Table 7 shows the amount of uranium and its enrichment per MBP at varying burnups.

Table 7: U per MBP

Burnup of 240 MWd/MT	
U per core (g)	50400000
Average ^{235}U Enrichment	0.00703
Average ^{235}U (g) per core	354000
Average ^{235}U per rod (g)	44.3
Average ^{235}U per batch (g)	886
Average ^{235}U per day (g)	1770
Average ^{235}U per MBP (g)	53100

The TransLAT output gives the amount of uranium in the 4x4 array, so those values have been extrapolated over the entire core. The MBP remains 30 days to achieve those values. At beginning of life in the core the right amount of uranium is present, so the values look appropriate. Also, the amount of uranium processed per day is around 250 kg, which is in the ballpark of the 375 kg that was stated in 1994 as their peak daily capacity.⁽²⁹⁾ The enrichment is slightly higher than the input file shows, but is nonetheless close to the true value at beginning of life. The rest of the values, including the later enrichment values, are average values derived in the same way that the average plutonium values were. The chemical processes will also cause uranium losses, and the same percentages will be used to estimate those losses.

Table 8: U losses

MBA #2 U	^{235}U (g)	Waste ^{235}U (g)
^{235}U after Decladding	52700	425
^{235}U after Dissolving	52600	79.1
^{235}U after FP Removal	52600	79
^{235}U after Purification	52500	52.6

Table 8 is calculated exactly the same way as Table 5, and gives the second mass to use for the uncertainty analysis. Using the same burnup correlation model from the plutonium cycle with the same uncertainties, the incoming σ_{MUF} can be calculated. For the outgoing uranium, a different system is needed. This will require a combination of systems, first the material will be weighed using an electronic balance (EBAL in the ITV-2000). This measurement should have a systematic and random uncertainty of 0.5%.⁽²⁷⁾ The material will also be measured using a portable NaI detector to get the enrichment, with a systematic and random uncertainty of 5%.⁽²⁷⁾ Table 9 has the anticipated results from the uranium uncertainty analysis.

Table 9: Uranium Measurement Uncertainty Analysis

M1 Sigma MUF ^{235}U (g)	13400
M2 Sigma MUF ^{235}U (g)	3710
Sigma MUF ^{235}U (g)	14000
IAEA target ^{235}U (g)	25000
MBP = 30 days	

As can be seen in Table 9, the M1 measurement has a high uncertainty associated with it. This drives the achievable MBP much lower, and so limits the allowable MBP for the whole facility. That result is surprising, because normally plutonium would be the overriding concern of any safeguards regime, however in this facility the uranium poses a more important factor when the timeliness goals are considered. Therefore, the MBP for the Radiochemical Laboratory must be set at 40 days to achieve all of the requirements of the timeliness goals.

The final check on the goals of the accountancy program shows that all four diversion pathways mentioned previously are detected. A missing spent fuel rod would be detected when the number leaving storage did not match the number entering storage.

The amount of plutonium is estimated before any fuel is dissolved, and the estimation is sufficient to detect a diversion of any of the dissolved fuel with or without fission products. The amount of plutonium available in the fuel also works in the favor of the accountancy program, because the plutonium is very dilute. Dilute plutonium would require a massive diversion to build up any amount of plutonium larger than gram size quantities. Diverting stored PuO_2 would also be detected because there is a very good measurement of the amount of plutonium entering storage. That plutonium is to be stored in cans and sealed. As long as the seal is not broken, the amount of plutonium in each can will remain the same and can be re-verified at any time.

CHAPTER IV

SUMMARY AND CONCLUSION

The first step in implementing safeguards on any fuel cycle is to understand what facilities are present in the fuel cycle and what their responsibilities to each other are. In the case of the DPRK, there are 3 major facilities: the Fuel Fabrication Facility, the 5 MWe Reactor, and the Radiochemical Laboratory. The Fuel Fabrication Facility was originally designed to fuel the 3 planned reactors, with a top capacity of 700 MT/yr of uranium metal fuel rods, however its current capacity is a smaller 25 MT/yr. The 5 MWe Reactor is the only completed production reactor and completes burning one core load of fuel every 2 years. The Radiochemical Laboratory was also designed to accommodate a larger capacity and reprocess the fuel of at least 2 of the planned reactors and has reprocessed 50 MT of spent fuel within 6 months in the past.

The goals of the safeguards system were set based off the significant quantity and timeliness goals that the IAEA uses. In order to achieve those goals, the 5 MWe Reactor had to be modeled to simulate its capabilities. This modeling must be complete, from estimating plutonium production to estimating the full thermal limits of the reactors. From open source references, the dimensions and characteristics of the reactor can be modeled in a computer code. There are several to choose from, for this case TransLAT was used. The TransLAT simulation estimated that roughly 6.7 kg of plutonium could be produced per year in the 5 MWe Reactor, which is in the upper range of the 6-7kg estimates from ISIS.⁽¹¹⁾ TransLAT also calculated averaged cross sections used to calculate the axial power profile in the reactor. The profile was solved using a two-group, two-region diffusion system of equations that verified the 20 MWth output of the core.

The throughput of material for each facility was researched. Each facility consists of processes that transform or form the special nuclear material. Open source material was used to generate the needed data. The process that occurs in the Fuel Fabrication Facility is fairly straightforward, and due to the constraints of monitoring

natural uranium, the key measurement points are easily identified as the input from the mine and the output of finished fuel rods. The Radiochemical Laboratory uses the PUREX process, which has been well understood for decades. Due to the presence of both uranium and plutonium, both materials need to be accounted for. For this approach, the spent fuel is analyzed and the plutonium content is actually estimated for use in the material accountancy. This is abnormal for a reprocessing facility, however the amount of material that is processed in this facility allows for the higher uncertainties associated with this measurement.

Proliferation pathways are present in every facility, and in order to maintain confidence in the safeguards regime they must be addressed. At the Fuel Fabrication Facility, the natural uranium used causes the potential proliferator to divert large amounts of the material to be effective. The 5 MWe Reactor is easy to safeguard thanks to the item accounting that can be conducted. Any missing rods are easy to notice, and a single rod will not provide enough material for a proliferator to fashion into a weapon. The Radiochemical Laboratory has several proliferation pathways, but is still fairly easy to safeguard because of the relatively low amounts of plutonium passing through the facility at any one time. Finally, an uncertainty analysis of the measurement systems was conducted. Each measurement has an uncertainty associated with it, and that uncertainty leads to MUF. By ensuring that the σ_{MUF} is below one third of the significant quantity of the material in question, there can be confidence in the integrity of the safeguards system. For the Fuel Fabrication Facility, the MBP can be as long as 365 days at its current capacity or as small as 14 days if the capacity is made larger to support a potential 200 MWe Reactor. The amount of material is vastly different for the two capacities, which is the cause of the different MBP's. The 5 MWe Reactor does not have an MBP due to containment and surveillance of the fuel rods. As long as each fuel rod is accounted for, it is unlikely for any material to be diverted. The Radiochemical Laboratory must have an MBP of no longer than 30 days.

Fuel cycles that feature small graphite-moderated and gas-cooled reactors are capable of implementing safeguards that meet the timeliness goals set by the IAEA with

minimal cost and minimal impact on facility operations. The test case shows that former weapons programs can also be adapted to suit the needs of a safeguards regime. Thus, any rejection of safeguards implementation by a state with this set of facilities suggests that this state has ulterior or nefarious motives.

This research also explicitly shows that it is possible to safeguard graphite moderated reactors and associated fuel cycle facilities and that the process is also very straightforward and similar to the process for LWR's. Specifically, the current fuel cycle present in the DPRK can also be safeguarded with minimal impact to the legitimate purposes of the fuel cycle. As has been shown, the front end of the fuel cycle can have a much higher capacity and still meet the goals set forth by the IAEA. The Radiochemical Laboratory is also capable of an increase in capacity without negatively affecting the facilities operation. Therefore, a larger reactor or additional reactors could also be included in this safeguards approach with only slight modification of the proposed system. The rejection of safeguards implementation for similar facilities would imply something about that state's intentions for the fuel cycle, and could be used to make a case for noncompliance with the NPT.

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APPENDIX A

TransLAT INPUT DECK WITH CONTROL ROD MATERIAL PRESENT

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TTL DPRK 5MWe fuel pin cell
con:0
TRP:D MOCS2D /
TRP:N MOCS2D /
TRP:G OFF
! Operating state
SYS PD=0.4 TF=900.0 TC=620.0 TM=600.0 TG=573.0 PR=2000
DOP On
! Mat Data
MAT:FUEL 18.9 92234=0.0054725 92235=0.7164 92238=98.7781275 13000=0.5
TAVE=TF MATTYP=1 ICHN=92/
MAT:CLAD 1.738 40000=0.5 13000=99.5 TAVE=TC MATTYP=0/
MAT:GRPH 1.2 6012=100 5000=7.5E-6 26000=309.9E-4 TAVE=TM MATTYP=0/
MAT:COOL 0 6012=7.69E-5 8016=1.54E-4 TAVE=TG MATTYP=0/
MAT:CAP 0 13000=0.0215 6012=3.84E-5 8016=7.69E-5 TAVE=TC MATTYP=0/
MAT:CTRL 2.52 5000=80 6012=20 TAVE=TG MATTYP=0/
!-Fuel Pieces
GEO:1
'REG',40/
'RPP','GRPH',1,1,10.0/
'RCC','GRPH',1,1,4.25/
'RCC','GRPH',1,1,3.9/
'RCC','GRPH',1,1,3.7/
'RCC','GRPH',1,1,3.6/
'RCC','GRPH',1,1,3.5/
'RCC','COOL',1,1,3.25/
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'RCC','CLAD',1,1,1.5/
'RCC','FUEL',1,1,1.45/
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'RCC','FUEL',1,1,1.435/
'RCC','FUEL',1,1,1.43/
'RCC','FUEL',1,1,1.425/
'RCC','FUEL',1,1,1.42/
'RCC','FUEL',1,1,1.415/
'RCC','FUEL',1,1,1.41/
'RCC','FUEL',1,1,1.405/
'RCC','FUEL',1,1,1.40/
'RCC','FUEL',1,1,1.39/
'RCC','FUEL',1,1,1.38/
'RCC','FUEL',1,1,1.37/
'RCC','FUEL',1,1,1.36/
'RCC','FUEL',1,1,1.35/
'RCC','FUEL',1,1,1.325/
'RCC','FUEL',1,1,1.30/
'RCC','FUEL',1,1,1.25/

```

```

'RCC','FUEL',1,1,1.20/
'RCC','FUEL',1,1,1.15/
'RCC','FUEL',1,1,1.10/
'RCC','FUEL',1,1,1.05/
'RCC','FUEL',1,1,1.00/
'RCC','FUEL',1,1,0.90/
'RCC','FUEL',1,1,0.80/
'RCC','FUEL',1,1,0.70/
'RCC','FUEL',1,1,0.60/
'RCC','FUEL',1,1,0.50/
'RCC','FUEL',1,1,0.40/
'RCC','FUEL',1,1,0.30/
'CTRLF',40/
'RPP','GRPH',1,1,4.5/
'RCC','GRPH',1,1,4.25/
'RCC','GRPH',1,1,3.9/
'RCC','GRPH',1,1,3.7/
'RCC','GRPH',1,1,3.6/
'RCC','GRPH',1,1,3.5/
'RCC','COOL',1,1,3.25/
'RCC','CAP',1,1,2.5/
'RCC','CLAD',1,1,1.5/
'RCC','FUEL',1,1,1.45/
'RCC','FUEL',1,1,1.445/
'RCC','FUEL',1,1,1.44/
'RCC','FUEL',1,1,1.435/
'RCC','FUEL',1,1,1.43/
'RCC','FUEL',1,1,1.425/
'RCC','FUEL',1,1,1.42/
'RCC','FUEL',1,1,1.415/
'RCC','FUEL',1,1,1.41/
'RCC','FUEL',1,1,1.405/
'RCC','FUEL',1,1,1.40/
'RCC','FUEL',1,1,1.39/
'RCC','FUEL',1,1,1.38/
'RCC','FUEL',1,1,1.37/
'RCC','FUEL',1,1,1.36/
'RCC','FUEL',1,1,1.35/
'RCC','FUEL',1,1,1.325/
'RCC','FUEL',1,1,1.30/
'RCC','FUEL',1,1,1.25/
'RCC','FUEL',1,1,1.20/
'RCC','FUEL',1,1,1.15/
'RCC','FUEL',1,1,1.10/
'RCC','FUEL',1,1,1.05/
'RCC','FUEL',1,1,1.00/
'RCC','FUEL',1,1,0.90/
'RCC','FUEL',1,1,0.80/
'RCC','FUEL',1,1,0.70/
'RCC','FUEL',1,1,0.60/
'RCC','FUEL',1,1,0.50/
'RCC','FUEL',1,1,0.40/
'RCC','FUEL',1,1,0.30/
'CTRLR',2,0,1,0,1,0/

```

```

'RPP','GRPH',1,1,5.5/
'RCC','COOL',1,1,3.25/
/
!-Graphite Blocks
GEO:41
'Long','GRPH',14.5,5.5/
'NS','GRPH',5.5,14.5/
'Short','GRPH',5.5,9.0/
/
LAT
+REG:1-12/
  4*0.0 20,3*0.0 0.0,20,2*0.0 40,3*0.0 60,3*0.0 0.0,40,2*0.0
0.0,60,2*0.0 20,60,2*0.0 40,60,2*0.0 60,20,2*0.0 60,40,2*0.0
60,60,2*0.0/
+CTRLF:1-4/
  20,20,2*0.0 40,20,2*0.0 20,40,2*0.0 40,40,2*0.0/
+CTRLR:1/
  30,30,0.0,0.0/
+Long:1-8/
  15.5,10,2*0.0 30,10,2*0.0 10,24.5,2*0.0 35.5,24.5,2*0.0
10,30,2*0.0 35.5,30,2*0.0 15.5,44.5,2*0.0 30,44.5,2*0.0/
+NS:1-4/
  10,10,2*0.0 44.5,10,2*0.0 10,35.5,2*0.0 44.5,35.5,2*0.0/
+Short:1-4/
  24.5,15.5,2*0.0 24.5,35.5,2*0.0 30,15.5,2*0.0 30,35.5,2*0.0/
/
! Print
PRI
  1/
  20,2,1,1,1,1/
  30,1/
/
BUR
1,2,3,4,4.5,4.8,4.81,4.82,4.83,4.84,4.85,4.86,4.87 /
! MCNP
TRA
  1,1,1000,5 2000,1.3, '5mwe', '/packages/translat/tfx/tramcnp_vi.dat'/
!
STA

```


APPENDIX B

MAPLE WORKSHEET USED TO FIND POWER PROFILE

```

> sigmaa1c:=7.83606E-4;sigmaa2c:=2.40453E-
3;sigmaa1r:=1.16939E-3;sigmaa2r:=3.0463E-
3;sigmatr1r:=2.34276E-1;sigmatr2r:=1.544E-
1;sigmatr1c:=2.31652E-1;sigmatr2c:=1.52268E-
1;nusigf1c:=4.10675E-4;nusigf2c:=2.84748E-
3;nusigf1r:=4.07371E-4;nusigf2r:=2.85055E-
3;d1c:=1/(3*sigmatr1c);d2c:=1/(3*sigmatr2c);d1r:=1/(3*sigma
tr1r);d2r:=1/(3*sigmatr2r);Bm11:=(nusigf1c-
sigmaa1c)/d1c;Bm21:=(nusigf2c-
sigmaa2c)/d2c;Bm12:=(nusigf1r-
sigmaa1r)/d1r;Bm22:=(nusigf2r-
sigmaa2r)/d2r;sigmar1c:=2.42349E-3;sigmas1t2c:=2.42349E-3-
7.83606E-4;sigmar1r:=2.41823E-3;sigmas1t2r:=2.41823E-3-
1.16939E-3;sigf2c:=1.16922E-3;sigf1c:=1.56738E-
4;sigf1r:=1.55613E-4;sigf2r:=1.17048E-3;
> L:=590;a:=30;
> c11:=sigmar1c-
nusigf1c;c12:=1/nusigf2c;c13:=c11/d1c+sigmaa2c/d2c;c14:=(si
gmas1t2c-sigmaa2c*c11*c12)/(d1c*d2c*c12);
> mu21:=(1/2)*(-
c13+(c13^2+4*c14)^(1/2));lambda21:=(1/2)*(c13+(c13^2+4*c14)
^(1/2));
>
c15:=sigmas1t2c/(d2c*mu21+sigmaa2c);c16:=sigmas1t2c/(sigmaa
2c-d2c*lambda21);
> phi1c:=x-
>A*cos(sqrt(mu21)*x)+C*cosh(sqrt(lambda21)*x);phi2c:=x-
>A*c15*cos(sqrt(mu21)*x)+C*c16*cosh(sqrt(lambda21)*x);
> c21:=sigmar1r-
nusigf1r;c22:=1/nusigf2r;c23:=c21/d1r+sigmaa2r/d2r;c24:=(si
gmas1t2r-sigmaa2r*c21*c22)/(d1r*d2r*c22);
> mu22:=(1/2)*(-
c23+(c23^2+4*c24)^(1/2));lambda22:=(1/2)*(c23+(c23^2+4*c24)
^(1/2));
>
c25:=sigmas1t2r/(d2r*mu22+sigmaa2r);c26:=sigmas1t2r/(sigmaa
2r-d2r*lambda22);

```

```

> philr:=x-
>E*cos(sqrt(mu22)*x)+G*cosh(sqrt(lambda22)*x);phi2r:=x-
>E*c25*cos(sqrt(mu22)*x)+G*c26*cosh(sqrt(lambda22)*x);
> eq1:=philc(-L/2-2*d1c)=0;eq2:=phi2c(-L/2-
2*d2c)=0;eq3:=philr(L/2+2*d1r)=0;eq4:=phi2r(L/2+2*d2r)=0;
> philc(x)>0;phi2c(x)>0;philr(x)>0;phi2r(x)>0;
> eq5:=philc(a)=philr(a);eq6:=phi2c(a)=phi2r(a);
> dphilc:=x->-
A*sqrt(mu21)*sin(sqrt(mu21)*x)+C*sqrt(lambda21)*sinh(sqrt(l
ambda21)*x);dphi2c:=x->-
A*c15*sqrt(mu21)*sin(sqrt(mu21)*x)+C*c16*sqrt(lambda21)*cos
h(sqrt(lambda21)*x);dphilr:=x->-
E*sqrt(mu22)*sin(sqrt(mu22)*x)+G*sqrt(lambda22)*sinh(sqrt(l
ambda22)*x);dphi2r:=x->-
E*c25*sqrt(mu22)*sin(sqrt(mu22)*x)+G*c26*sqrt(lambda22)*sin
h(sqrt(lambda22)*x);
> eq7:=-d1c*dphilc(a)=-d1r*dphilr(a);eq8:=-d2c*dphi2c(a)=-
d2r*dphi2r(a);
> solve({eq6,eq7,eq8},{C,E,G});
> G:=-1.527758747*A;C:=-1.431608855*A;E:=0.9648199562*A;
> Ef:=3.20435459999999996E-17;Pf:=20;
> eq9:=Pf=int(Ef*(sigf1c*philc(x)+sigf2c*phi2c(x)),x=-
L/2..a)+int(Ef*(sigf1r(x)*philr(x)+sigf2r*phi2r(x)),x=a..L/
2);
> solve(eq9,A);

```

VITA

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